R0257

FOURTH REPORT

OF THE

TMI-2 TECHNICAL ASSISTANCE AND ADVISORY GROUP (TAAG)

DECEMBER 1, 1982

TAAG MEMBERS

W. H. Hamilton, Chairman
R. Brodsky
D. O. Campbell
N. M. Cole/D. Strawson
T. S. Cramer/T. E. Peterson
E. A. Evans
E. F. Sise/R. McNamara
E. J. Wagner

PARTICIPATING

H. M. Burton, EG&G
F. A. Ross, DOE
R. Weller, NRC
A. Roberts, EPRI

Produced under
Head Commissioner
page 16

FOURTH REPORT OF TAAG

OUTLINE

FOURTH REPORT OF TAAG

I. INTRODUCTION

This report covers the TAAG activities during the period of time from September 1, 1982, to December 1, 1982. In letters addressed to W. H. Hamilton, Chairman, TAAG, by B. K. Kanga, Director, TMI-2, the following items were listed as topics for TAAG activity during the subject time interval:

(1) An in-depth evaluation of the fuel transfer tube carriage and associated mechanism connecting the Containment Building refueling canal with the Fuel Handling Building. These are the intended vehicle for transport of canned fuel from containment to the "A" fuel pool for interim storage. Babcock and Wilcox (B&W) has proposed certain modifications, based on operating experience, to improve the operability of the system. TAAG's review of the B&W system, and proposed modifications, is requested to provide assurance of the adequacy of that system for the entire defueling process.

As an associated evaluation, TAAG should assess timing and sequence of this required modification work, taking into account the potential need to flood the refueling canal for head lift, as well as potential scheduling interferences on the Fuel Handling Building end.

(2) An evaluation of the potential for presence of hydrides (not only zirconium hydride but any hydride forms) which could cause a severe exothermic reaction. TAAG's evaluation on this subject should consider potential forms and locations, conditions required for ignition, and ramifications on the defueling plan.

(3) Follow and evaluate the effectiveness of the reactor building decontamination and characterization.

(4) Assess the plans for plenum removal and underhead examination.

(5) Follow the items listed by TAAG as pre-requisites for head removal.

(6) Assess the plans for modifying the SDS system and for processing canal water.

(7) Appraise the man-rem estimates in the PEIS.

(8) Follow preparation of an interface document for design criteria on canisters by EG&G/OCE/GPU and evaluate the plans for the canister program.

The report is organized by having a section for each of the above subjects. Conclusions and recommendations are included in each section. Where the work is not yet completed on the subject, a status report of work to date is included and a statement made regarding any plans for future TAAG effort.

FUEL CONTENT IN REACTOR COOLANT SYSTEM

This item is reported as a follow-up to prior TAAG actions and reports.

Estimates of fuel debris in the RCS vary from a few kilograms to many tons. To decrease the uncertainty a limited determination of debris in the steam generators was recommended by TAAG. The action is planned during the next quarter when the water level is again lowered. Meanwhile, two techniques considered have been used recently to locate fuel in a TMI-2 demineralizer. These results are summarized below; more details are included in Reference 1.

Si(Li) GAMMA-RAY DOSIMETRY RESULTS

During October 1982, gamma spectrometry measurements were carried out to assess the fuel debris content of the TMI-2 makeup demineralizers. A shielded Si(Li) Compton recoil gamma ray spectrometer was used to measure the gamma spectra at various locations within the cell. The spectral data were used to determine the intensity of the 2.18 MeV gamma ray from the fission product ^{144}Ce. Assuming this fission product does not migrate out of the fuel, the quantity of ^{144}Ce is directly related to the quantity of fuel present.

These spectral measurements also provided data for determining the ^{137}Cs loading on the demineralizer resin. There are gross nonuniformities in the source distribution. Based on the observed source geometry and the measured flux of the ^{144}Ce 2.18 MeV gamma-rays, the fuel content of the A demineralizer is calculated to be 1.3 $\pm$ 0.6 Kg. In addition, as based on these measurements, the ^{137}Cs content is calculated to be 3400 $\pm$ 2500 Curies. Both estimates are as of mid-October, 1982.

SOLID STATE TRACK RECORDER (SSTR) NEUTRON DOSIMETRY RESULTS

An adequate signal has been obtained from a 30 day SSTR exposure on TMI-2 demineralizer A. SSTR results complement and generally agree with Si(Li) gamma-ray dosimetry results. These neutron data confirm that the demineralizer A tank is essentially empty above the 309' elevation. Using a point source assumption, the SSTR provides bounds for the fuel content in demineralizer A as follows: a 0.5 kg lower limit and a 20 kg upper limit. However, the most reliable estimate to date is based on "room return" thermal neutrons, which provides a result of ~3 kg.

Reference 1. Fuel content of the Three Mile Island Unit 2 Makeup Demineralizer, (In Publication) James P. McNeece, ETAL., HEDL, Dec. 1982.

II. FUEL TRANSFER MECHANISM

TAAG was requested by GPU/Bechtel (Kanga letter dated September 15, 1982) to review the fuel transfer mechanism, between the Reactor Building and Fuel Pool, and the proposed modifications to determine reliability. There are two such mechanisms at TMI-2 which must be very reliable in their operation. A breakdown of one or both would cause a serious impact to the defueling program.

On September 24, 1982 TAAG members were provided a review package, consisting of technical reports and drawings of the existing mechanism and a proposed concept for the modification. During the TAAG meeting of October 14, 1982 an extensive review of a proposed modification concept was held with GPU, Bechtel, Babcock & Wilcox, and Sterns & Rogers personnel. The existing mechanisms in general consist of a rail system, transfer tube, carriage/basket assembly, air motor drive, sprocket and chain, and upenders. The modifications proposed are: replace the chain and sprocket drive with a single cable/drum/pulley drive; replace existing carriage bushings with lubrite bushings and remove limit switches from carriage operations; and install carriage drive and controls on the edge of the fuel pool out of the water.

At completion of the review, TAAG had the following comments:

1. During the review session it became apparent that the review package was conceptual only and not yet complete in Engineering or design.

2. That a review of all past problems experienced with the transfer mechanism be conducted and that these problems be addressed during the proposed modifications.

3. Along with the replacement of the carriage drive, bushings in other portions of the system (i.e., upenders) should be evaluated for replacement.

4. Boron concentration (~3600 PPM) in the fuel cavity will be higher than previously experienced by a fuel transfer system and chemicals may be added for clarity, pH, etc. All of this must be considered in selection of equipment and material for proper operation of the whole fuel transfer system (i.e., upender hydraulic piston shafts, seals, etc.).

5. Asymetrical loads were not considered in the redesign of the fuel transfer mechanism; in fact, the latch system is designed for a symetrically loaded fuel cell only. This must be taken into account.

6. A thorough detailed inspection of the fuel transfer mechanism equipment that is intended to be reused must be performed to insure equipment repairs/replacements are identified in advance.

7. Removal of the limit switch for control of movement of the carriage assembly is part of the proposed modifications. This action is based on past unreliable performance of these limit switches. It is recommended that limit switch use in the remainder of the system be evaluated on the same basis and that limit switch set points be established at approximately 150% of expected operating load instead of system failure point. Further, the limit switch mountings and their associated wiring should be modified to be readily removable from above the pool water surface.

8. The transfer tube flange covers in the Reactor Building should be removed prior to flooding the canal or be capable of being removed remotely in case of flooding of canal. The only method at present for removal, when the canal is flooded, is for a diver to go down and manually remove the flange covers.

9. Required inspections and modifications to those parts of the transfer mechanism located in the Reactor Building should be completed prior to head lift. If the canal is flooded for head or plenum removal, a considerable amount of time, money and man-rem exposure would be expended to drain the canal and decontaminate the canal and equipment in order to perform the inspections and modifications after the canal had been flooded.

10. Transfer mechanism rails are supported on the fuel cavity floor by embedments. A question was raised as to embedment integrity, causing misalignment when the canal is partially filled. This area should be fully evaluated.

TAAG recommends that the comments above be evaluated and incorporated into the final engineering and design package. TAAG also requests that, after completion of the final package, another TAAG review be conducted.

III. PYROPHORICITY OF MATERIALS IN THE REACTOR

TAAG was asked to make an evaluation of the potential for presence of any material such as zirconium metal or hydride which could cause a severe exothermic reaction. TAAG's evaluation on this subject was to consider potential forms and locations, conditions required for ignition, and ramifications on the defueling plan.

In addressing the matter of pyrophoricity of fuel material, TAAG called in people for advice on this matter:

J. D. Watrous, Westinghouse Hanford Company
M. L. Picklesimer, NRC (retired)
D. E. Owens, EG&G Idaho
H. M. Chung, Argonne National Lab.

In addition, a literature search was made. Important documents used are listed in the table of references at the end of this section of the report. Coupled with the technological aspects of the pyrophoricity of fuel material were detailed examinations of methods planned for reactor component removal.

Pyrophoricity of Zirconium and Zirconium - Hydrides

It has long been known that many metallic powders, including zirconium metal and hydrides are hazardous to handle in air. This is due to their relative ease of ignition and the large amount and high rate of energy released during combustion. There is also substantial experience that partially moist (e.g., up to 25% moisture) zirconium powders are particularly hazardous to handle because of the reaction with water once ignition occurs.

A number of theories have been developed to explain the various fires and explosions that have occurred since zirconium has been used for nuclear reactors. It is known that the pyrophoric tendency is greater with particles of increasing surface-to-volume ratios. That is, powder particles in the micron range appear to be the most likely to ignite. Larger particles, however, such as machine turnings can also ignite and burn. In reference 1

written in 1956, nine types of mechanisms that lead to the pyrophoricity of zirconium are discussed. Today there is still not a generally accepted theory or explanation of the fires and explosions which have occurred. Furthermore, the occurence of such fires is sometimes random and unpredictable. This being the case, the approach has been to exercise extra caution in handling the material. The National Safety Council published in 1974 a data sheet which lists the considerations in containing, shipping, storage and handling of zirconiums material. That data sheet can be used in future design work for the handling of TMI-2 fuel debris.

In addition to the tendency for unoxidized metallic zirconium to auto-ignite, it has also been shown that zirconium hydride is pyrophoric. Hydrogen is taken up by zirconium in the presence of steam or moisture which is heavily saturated with hydrogen. The hydride is normally in the form of needles or platelets in the zirconium metal. The uptake of hydrogen is believed to be impeded by a corrosion film on the surface of the metal.

Under the abnormal conditions during the TMI-2 accident, however, formation of bulk hydride in localized regions of the core cannot be excluded. For example, in the case of a fuel rod such as those in TMI-2 in which the cladding was ruptured and steam entered inside the zirconium-clad fuel rod, there is a strong likelihood of hydriding on the ID of the cladding since there is no protective corrosion film. This is in contrast to the outside of the cladding which has an adherent protective corrosion film.

Core Debris in TMI-2

"A listing of core materials (shown below) results in only one major source for pyrophoric reactions - the zircalloy 4. The metallic Inconel 719, stainless steel, Ag-In-Cd poison and Be-Ni5 braze alloy either melted, reacted with each other or probably oxidized to a stable condition. The ceramic UO_2 fuel, $Al_2O_3 - B_4C$ and $UO_2 - Gd_2O_3$ poisons either remained intact, fragmented due to thermal shock or partially reacted with metallic

materials, resulting in thermodynamically stable materials at or near room temperature. Fission products are both less hazardous and in smaller quantities than zircaloy reaction products insofar as pyrophoricity is concerned."

"Products of the zircaloy-4 components include the undamaged cladding and guide tubes, hydrided intact components, powdered zirconium hydride, powdered zirconium oxide, fragmented zircalloy and reaction products with other metallic and ceramic materials in the core. ----" (See Reference 2)

CORE MATERIALS

UO_2

Zircaloy-4 (Fuel Rod Cladding and Guide Tubes)

ZrO_2 (By-Product Metal-Water Reaction)

Inconel 718 (Spacer Grid)

Ag-In-Cd (Control Rod Poison)

304 SS (Cladding of Control Rods and Axial Axial Power Shape Rods)

SS, Grade CF-3M

Al_2O_3-B_4C (Axial Power Shape Rods)

Uo_2-Gd_2O_3 (2 Fuel Assemblies Contained Gadolinia Test Rods)

Be-Ni5 (Braze for Control Rod Guides to Structural Support Plates in Upper Plenum)

In addition to the materials listed in the table of core materials there may have been formed eutectic alloys containing silver, cadium and indium, zirconium, uranium, oxygen, and possibly iron and other metals which can have pyrophoric properties. But zircaloy and zirconium hydride materials are the major ones of concern. This has been confirmed by others - see references 3, 7 and 8. As the core is disassembled and exposed to air, there is a possibility that these materials could ignite. In the reactor vessel the core material will be covered with water, eliminating the possibility of igniting. In the vacuum system used with the canisters there may not be water and hence precautions will be required.

Head Removal

In Section V-B of this report an explanation is made regarding the likelihood of core debris in the under-head region or on top of the plenum assembly. To confirm the absence of core debris under-the-head radiation measurements are suggested. Analysis of any core particles on the lead screw may also confirm the presence of core debris in the under-head region. Also, analysis of the material on the surfaces of the threaded portions of sample section taken from the 8-H leadscrew should indicate whether there are any fine pieces of unoxidized zircaloy and if they are in sufficient concentration to cause a concern with pyrophoricity. The three sample sections from the 8-H leadscrew were from the same general elevation as the upper plenum assembly cover plate and up to the elevation of bottom of the leadscrew support tube.

Unless results from the radiation survey and lead screw material analysis indicate to the contrary, the dry head removal method, currently planned by GPU/Bechtel, can proceed on the basis of no special precautions for pyrophoricity. It is the TAAG estimate that the planned tests will be confirmatory and that today's planning can proceed on the basis of no special precautions for pyrophoricity.

Since the Safety Evaluation Report for Head Removal will, of necessity, address the potential safety issues of pyrophoricity, it may be desirable to determine whether the consequences of a pyrophoric reaction on the surface of the plenum would be serious. TAAG is investigating methods for conducting such an evaluation.

Plenum Removal

From the Quick Look tapes it is known that there is core debris on the lower support plates in the plenum assembly. Analysis of the tapes showed debris on plates four through ten. In addition, the tapes showed the remains of end fittings attached to the bottom grid plate of the plenum.

It is the current GPU/Bechtel plan to keep the plenum under water until it is hoisted above the indexing fixture. It is also the plan to sweep or vacuum the support plates prior to removal from the water and to remove material attached to the bottom surface of the plenum.

If these actions are completed and if no negative results are obtained in the two tests described above under "Head Removal", planning for plenum removal can proceed with no special precautions regarding pyrophoricity of the core material.

Core Removal and Disposal

Core disassembly in the reactor vessel is expected to be under water. But, in the event of chipping or cutting, new shapes of material will be made, which will have fresh unoxidized surfaces. Such zircaloy or zirconium hydride metal, when exposed to air, may be pyrophoric.

The in-reactor operations, so long as the work region is fully flooded, will not require other special precautions. But as the material is removed and canned, precautions probably will be required. The nature and scope of the precautions will be influenced by samples taken after head and plenum removal. But, it is currently anticipated that keeping the material covered with water or with an inert gas will be required.

Plans for core material removal, storage and shipment are at a very early stage. But, at this time until further evidence is developed, the plans should provide for protecting against possible pyrophoricity of the core materials.

REFERENCES

1. U. S. Atomic Energy Commission, Accident and Fire Prevention Information, Issue No. 45, "Zirconium Fire and Explosion Hazard Evaluation", August 7, 1956

2. "Pyrophoric Reaction Possibilities in TMI-2 Core", J. D. Watrous, Westinghouse Hanford Co., September 23, 1982.*

3. Discussion of the Possibility of the Formation of Zirconium Hydrides During the TMI-2 Accident and the Subsequent Possibility of Autoignition during Plant Clean-up", H. M. Chung, C. H. Bowees, Argonne National Laboratory, January 16, 1981.*

4. "Final Report of Explosion in Oak Ridge, Tennessee Y-12 Salvage Yard Adjacent to Building 9929-1", Union Carbide Nuclear Company, July 11, 1956.

5. "Explosibility of Metal Powders" by Murray Jacobson, Austin R. Cooper and John Nagay, U.S. Bureau of Mines, 1964.

6. "Fuel and Cladding Structures Formed During Severe High Temperature Transients", D. E. Owens, EG&G, Idaho, September, 1982.*

7. "Hydriding and Pyrophoricity in the TMI-2 Primary System", M. L. Picklesimer, September 1982.*

8. "EG&G Idaho Comments on Pyrophoric Materials in the TMI-2 Core Debris", D. E. Owens, EG&G Idaho, September, 1982.*

*Documents distributed to TAAG at the meeting held on this subject on September 23, 1982.

IV. REACTOR BUILDING DECONTAMINATION AND RADIOLOGICAL CHARACTERIZATION

A. Background

TAAG reports of May 15, 1982 and August 31, 1982, addressed the radiological conditions in the reactor building. Recommendations of these reports included:

a. Use of the pathway concept for operations during the period from reactor vessel head lift through fuel removal. The pathways would be equipped with contamination enclosures (tents), air filtration and shielding as necessary to control gamma dose rates, airborne and surface contamination.

b. Identification of sources contributing to the gamma dose rates above the EL 305' floor.

c. Recognition of the difficulty of decontaminating coated and uncoated concrete. In particular the surfaces of the reactor building that were flooded are expected to be significant radiation sources from Cs^{137} that penetrated the concrete.

d. Observation of the leaching of Cs^{137} and Sr^{90} from materials in the EL 282'6". Leaching, if it occurs to a substantial extent, could be utilized to effect some decontamination of the wetted surfaces of EL 282'6".

During this period TAAG reviewed the reactor building decontamination and radiological conditions in four areas:

a. Plan for radiation characterization of the reactor building
b. Plan for gamma dose rate reductions
c. Results of ongoing reactor building decontamination
d. Decontamination of concrete

This section provides TAAG's conclusions from these reviews and makes specific recommendations concerning radiological engineering to support defueling.

B. Conclusion

Efforts to date to establish a satisfactory radiological environment in the reactor building have produced limited success. Reduction of smearable surface contamination has initially met targets but frequently showed recontamination; airborne particulate concentrations have been substantially reduced but not below respiratory protection limits; and gamma dose rates have not been reduced substantially. Cesium 134 and 137 are the principal contaminants in all modes. Indications are present of penetrations of cesium into paint on metal surfaces, into coatings on concrete surfaces and into various types of concrete in the reactor building. The penetration of cesium appears to provide a subsurface reservoir of contaminant to re-contaminate surfaces once decontaminated, supply a continuous source of airborne contaminant and generate the thus-far irreducible gamma dose rates.

The initial planning of actions to establish a satisfactory radiological environment treated the reactor building with a monoblock view. The whole building was viewed as the work area and actions, such as the gross decontamination, were planned to apply to the whole building. While it is a forgone conclusion that the whole building must eventually be decontaminated, the difficulties experienced to date emphasize the need to concentrate effort on establishing satisfactory radiological conditions in specific work areas and access pathways.

The priority activity is defueling. TAAG concludes that the radiological engineering needs to be focused on support of those operations contributing to defueling (e.g., reactor vessel head lift, plenum removal, refueling canal preparations) rather than in generally improving the radiological environment throughout the reactor building. Considerations of specific facets of the radiological engineering and TAAG action recommendations follow.

C. Dose Rate Reduction

GPU/Bechtel presented the preliminary results of a task force effort toward determining radiation sources and defining a gamma dose rate reduction program. The results were based upon analyses of dose rate measurements by TLD's suspended between the EL 305' and EL 282'6" floors at various locations. These analyses identified the significant gamma sources and their contribution to dose rates above the EL 305' floor. A program of dose rate reduction was defined and GPU/Bechtel advised that the program is being implemented promptly.

The program is summarized in Attachment IV-1. TAAG applauds the efforts of this task force and endorses the program they developed. We note that it incorporates a broad range of the techniques of radiological engineering and extends beyond the specific area of gamma dose rate reductions. This is an important step to a comprehensive radiological engineering program in contrast to separate planning for decontamination, airborne activity control, dose rate reduction, or water cleanup.

TAAG recommends the following actions relative to this program:

a. Develop a radiological engineering plan to support each principal work activity, such as head lift, plenum removal and fuel removal. Include all the techniques available for establishing a satisfactory radiological environment for the performance of each principal work activity.

b. Direct the priorities of the radiological engineering actions to support specific principal work. At present the principal work is head lift, plenum removal and fuel removal. Efforts to reduce dose rates in areas not traveled for these activities should be deferred until needed to support the other work. Subsequent principal work efforts, such as the fuel debris search and removal, reactor vessel head refurbishment or reactor plenum removal from the reactor building should also have specific radiological engineering plans.

c. For each principal activity identify the work area and pathways to which radiological engineering actions need be taken. It is noted that TAAG recommended a specific pathway approach in their August 31, 1982, report. This recommendation is reiterated. (Note that Attachment IV-A-1 to the August 31, 1982, report described a specific pathway concept for head lift, plenum removal and fuel removal)

d. The characterization of radiological conditions be continued for work areas and pathways in EL 347' and 305'. For example, further paint coating and concrete samples should be taken in these areas to define the radiological conditions and applicable techniques for correction.

TAAG also reviewed certain aspects of the radiological conditions in the reactor building and offers recommendations concerning them. These recommendations may affect the radiological engineering actions discussed above and should be included among them at the appropriate time.

O. INTRUSION OF RADIOACTIVITY INTO AND LEACHING FROM CONCRETE

Various experiences, including the decontamination of the TMI-2 auxiliary building, indicates that cesium penetrates into coated and uncoated concrete. The cesium that enters the concrete also leaches out of the concrete under certain conditions. This effect can be used to some extent as a practical concrete decontamination process. The same effect causes re-contamination of surfaces after attempts at surface decontamination. A surface once decontaminated to low smearable activity will become re-contaminated by cesium leaching out into the surface from the interior of the concrete. Attachment IV-2 describes some of the experience and relates it to TMI-2 reactor building conditions. The GPU/Bechtel task force on dose rate reduction similarly showed the high degree of contamination of concrete surfaces.

Leaching must therefore be contended with in all contaminated concrete. The worst situation is expected to be the long-flooded concrete in EL 282'-6". A positive use of the leaching could be to remove contamination from EL 282'-6" by continually flooding concrete surfaces and removing the activity by processing through SDS. Attachment IV-3 is an evaluation of the reactor building sump water cesium and strontium concentrations to determine the rates of leaching from materials in the EL 282'6" area.

In summary Attachment IV-3 indicates that leaching is occuring. Data are available from only the last two pumpouts that is meaningful to leach rate. Using these data an estimate of the time constant for cesium leaching was derived. The constant was 127 days. This long time constant would indicate that leaching rate is too slow to use leaching for decontamination. Also all radiological conditions (surface contamination, airborne, gamma dose rates) may continue for long periods to be affected by the slow migration of subsurface cesium to the water or air environment of the concrete.

TAAG recommends that:

a. Reactor building fill/pumpout data continue to be evaluated to determine the characteristics of leaching (leach rates, water level effect, etc.). The purpose of this is to provide the basic data to show whether any effective cesium removal can be effected by repeated fill/pumpouts and whether leach rates will require availability of the SDS for sump water processing.

E. TREATMENT OF SLUDGE ON EL 282'6" FLOOR

One of the dose rate reduction actions in Attachment IV-1 is the removal of sludge from EL 282'6". It is believed by GPU/Bechtel to be a substantial source contribution to dose rates above the EL 305' floor.

In the preceeding discussion on leaching the focus was on the contaminants in the concrete. The sludge is also a reservoir of cesium. Leaching from the sludge is also discussed in Attachment IV-3. In the sludge sample of June 1982, 91% of the cesium in the total sample was either in the water fraction or the washings of the solids. Less than 9% remained in the solids fraction of the sludge. TAAG recommends that:

a. The leaching evaluation recommended above be augmented by taking additional sludge samples as the fill/pumpout cycles continue. The results will assist in determining the priority of sludge removal versus continued leaching to remove the radioactivity from the sludge.

b. During the fill/pumpout cycles evaluate the effect of water shielding on dose rates in the work areas. A factor of 10 reduction in gamma dose rate above the EL 305' floor should result from shielding the sludge and floor concrete sources with 15 to 18 inches of water (about 100,000 gals.). The results should provide guidance for the priority of physical sludge removal or the decision to use water shielding for dose rate reduction. Water shielding could be particulary effective from an ALARA standpoint for dose rate reduction above the thin portions of the El. 305' floor.

F. REDUCTION OF AIRBORNE PARTICULATE CONCENTRATIONS

The March 1982 gross decontamination experiment showed airborne particulate reductions of about a factor of 10 based on breathing zone apparatus (BZA) data. Current gross decontamination washdowns since September 1982 also show reductions, but the concentrations have not been reduced below the respiratory protection limits. Attachment IV-4 shows the BZA data for all entries. It should be noted that the major component of the airborne particulate activity is Cs^{137}.

The reaction of the airborne activity to decontamination may be showing a minimum airborne concentration in equilibrium with the cesium saturated paint, coating and concrete surfaces of the reactor building. The cesium could be migrating from subsurface locations into the air analogously to its leaching out of materials to the sump water. The existence of such an equilibrium could have substantial effect on the process of reduction of airborne particulate concentrations and elimination of the general need for personnel respiratory protection.

TAAG recommends that:

a. Samples of paint and coatings be taken and expeditiously analyzed to confirm the data obtained in prior SAI and EG&G sampling. Determine whether the cesium has saturated these materials.

b. Use the sample data as guidance in further radiological engineering actions toward reduction of airborne particulate concentrations. For example, aggressive paint stripping or paint overcoats may be indicated.

c. The effectiveness of increased, continuous reactor building purge filtration flow for airborne particulate reduction should be evaluated. This is an extention of a TAAG recommendation in the August 14, 1982 report. It suggested maximizing purge filtration flow using both the reactor building purge system and the auxiliary building supplementary air filtration systems. This evaluation should include measurement of airborne concentrations with two purge trains operating continuously for about one month. Should increased air flow reduce airborne concentrations, use of the auxiliary building supplementary filtration system should be considered.

ATTACHMENT IV-1

DOSE REDUCTION TASK FORCE

TMI UNIT II

SIGNIFICANT GAMMA SOURCE TERMS

1. EL. 282
   - FLOOR
   - WALLS
2. AIR COOLERS
3. ELEVATOR SHAFT AND ENCLOSED STAIRWELL
4. HEAD SERVICE STRUCTURE
5. DISCRETE SOURCES
   - FLOOR DRAINS
   - TRASH
   - LOCA DUCTS
   - RESIN COLUMN
   - WELDING MACHINE
   - CFT A/B
   - SEAL TABLE
   - POLAR CRANE COMPONENTS
6. SURFACE CONTAMINATION
7. REACTOR COOLANT SYSTEM

DOSE REDUCTION TASK FORCE

TMI UNIT II

DOSE REDUCTION RECOMMENDATIONS

PHASE 1

A. TECHNIQUES

1. PERSONNEL MANAGEMENT
   * OPEN INNER/OUTER PERS AIRLOCK DOORS
   * MODIFY INGRESS/EGRESS PATHS
2. TREAT DISCRETE SOURCES
3. FLUSH ELEVATOR AND ENCLOSED STAIRWELL
4. SHIELD ON 305' ELEVATION
   * COVERED HATCH
   * OPEN STAIRWELL
   * ELEV AND ENCLOSED STAIRWELL
5. PARTIAL DECON OF AIR COOLERS

B. ESTIMATED DOSE REDUCTION

1. REDUCE TRANSIT DOSE TO < 25 mREM
2. REDUCE POLAR CRANE DOSE RATES TO < 80 mREM/HR
3. REDUCE EL. 347' DOSE RATES TO < 100 mREM/HR

C. COMPLETION BY END OF 1ST QUARTER OF 1983

D. PHASE II PREREQUISITES

· CONTINUE ELEVATION 282' CHARACTERIZATION

DOSE REDUCTION TASK FORCE

TMI UNIT II

DOSE REDUCTION RECOMMENDATIONS

PHASE 1

DISCRETE SOURCES	TREATMENT
FLOOR DRAINS	SHIELD
TRASH	REMOVAL
RESIN COLUMN	ELUTE AND REMOVE
WELDING MACHINES	SHIELD AND REMOVE
CFT A/B DISCHARGE LINES	SHIELD
SEAL TABLE	DECON (SHIELD, IF REQUIRED)
POLAR CRANE COMPONENTS	SHIELD AS NECESSARY

DOSE REDUCTION TASK FORCE

TMI UNIT II

DOSE REDUCTION RECOMMENDATIONS

PHASE 2

A. TECHNIQUES

1. CONTINUE REACTOR BUILDING DECON PROGRAM
   OBJECTIVE TO REMOVE RESPIRATORS
2. DECON AND SHIELD Rx SERVICE STRUCTURE (2^{ND} QUARTER)
3. REDUCE ELEVATION 282'6" CONTRIBUTION (UP TO 18 MONTHS)
   * REMOVE SLUDGE
   * AGGRESSIVE DECON WALLS AND FLOOR
   * COAT/SCARIFY WALLS AND FLOOR
4. DECON D-RING INTERIORS AND EQUIPMENT SURFACES (1^{ST} QUARTER)
5. PROCESS RCS

B. ESTIMATED DOSE REDUCTION

1. REDUCE TRANSIT DOSE TO 10 mREM
2. REDUCE SERVICE STRUCTURE DOSE REATE TO 150 mREM/HR
3. REDUCE EL. 305' DOSE RATES TO < 100 mREM/HR

DOSE REDUCTION TASK FORCE

TMI UNIT II

DOSE REDUCTION RECOMMENDATIONS

PHASE 3

A. TECHNIQUES

1. CONTINUE REACTOR BUILDING DECON PROGRAM
   OBJECTIVE TO REMOVE RESPIRATORS
2. REMOVE SHIELDING ON ELEVATION 305'
3. IDENTIFY AND SHIELD HOT SPOTS
4. DECON AND/OR REMOVE AIR COOLERS
5. DECON DRAIN SYSTEM
   ALL ELEVATIONS
6. DECON OF PRIMARY SYSTEM COMPONENTS

B. PROJECTED ACHIEVEMENT

1. REDUCE DOSE RATES TO COMPLETION CRITERIA LEVELS
2. ELIMINATE NEED FOR RESPIRATORY PROTECTIVE EQUIPMENT

DOSE REDUCTION TASK FORCE

TMI UNIT II

REACTOR BUILDING DOSE RATES

	CURRENT	END PHASE 1	END PHASE 2
TRANSIT 347'/P.C. AIRLOCK	≅ 40 mREM	≅ 25 mREM	≅ 10 mREM
305'	≅ 350 mR/H	≅ 300 mR/H	< 100 mR/H
347'	≅ 150 mR/H	< 100 mR/H	≅ 50 mR/H
POLAR CRANE	≅ 120 mR/H	< 80 mR/H	≅ 50 mR/H
SERVICE STRUCTURE	≅ 600 mR/H	≅ 600 mR/H	≅ 150 mR/H
282'6"	5-300 R/H	5-300 R/H	?

ATTACHMENT IV-2

TECHNICAL MEMORANDUM

DATE 11/17/82

NUMBER
TM3680-3 (Rev. 2)

COPIES TO:
Oradell Library
Woodbury Library
Jacksonville Library
WWhite
ASDam-db
FJPatti
NLacy
RGagliardo
JMTuohy
RBurrow
VFricke
CWHess
pf 2 3680-01

TO E. J. Wagner

FROM C. W. Hess CW Hess

SUBJECT W.O. 3680-01
Technical Assistance & Advisory Group
Three Mile Island Unit 2 Recovery
Contamination Intrusion into Concrete

KEYWORDS TMI-2, Reactors, Contamination, Decomissioning

REFERENCES:

(1) "Gross Decontamination Experiment Report", Bechtel National, Inc., Nuclear Fuel Operations, June 1982, draft.

(2) "Three Mile Island Concrete Decontamination Experiance", A paper presented at the Concrete Decontamination Workshop by B. Irving of Viken Industries, Inc., September 1980.

(3) "Decontamination of Test Cell "C" at the Nuclear Rocket Development Station After a Reactor Accident", Los Alamos Scientific Laboratory, LA-3663-MS, January 1967.

(4) "Properties of Radioactive Wastes and Waste Containers; Progress Report No. 5", BNL-NUREG-50763, P. Colombo & R. M. Neilson, Jr., dated Dec. 1977.

(5) B&R Memorandum, "Porous Concrete Structures in Reactor Building Basement", C.W. Hess to E. J. Wagner, Rev. 1 dated 7/22/82.

(6) B&R letter, "Specification 2555-20, Production and Delivery of Concrete", January 22, 1971, Serial No. 909-UE from R. J. Dobbs to E.C. Nagle of UE&C.

(7) "Building Code Requirements for Reinforced Concrete" ACI Standard 318-77.

(8) "Standard Specification for Hollow Load-Bearing Concrete Masonry Units", American National Standard, ANSI/ASTM C90-75.

2

REFERENCES: 9) ORNL/TM 7448, published July 1980.

10) B&R Telecon, C. W. Hess (B&R) to L. Cacashca (BNL), "Concrete Permeability," 7/14/82.

11) B&R Telecon, C. W. Hess (B&R) to M. Fermin (BNL), "Concrete Ion Exchange Properties," 7/15/82.

12) B&R Telecon, C. W. Hess (B&R) to M. Fermin (BNL), "Concrete Ion Exchange Properties," 8/2/82.

13) B&R Calculation, "Dose Rates from TMI-2 Concrete," No. 3680-15-7, 4/29/82.

14) "Pre-and-Post-Decontamination Surface Deposition Data," Hmb-384-82, EG&G Idaho Inc., September 1982. 2

15) B&R Calculation, "Dose Rates from Water in TMI-2 Containment," No. 3680-15-8, 5/12/82.

16) B&R Calculation, "Area Dose Rates from TMI-2 Concrete," No. 3680-15-11, 8/5/82.

17) "Water Evolution from Heated Concrete," Hanford Engineering Development Laboratory, HEDL-TME78-87, February 1979.

ATTACHMENTS:
a) Table 1 - Cesium Source Terms for Unpainted Concrete, EL 282' 6"
b) Table 2 - Water Intrusion in TMI-2 Reactor Building Concrete
c) Table 3 - Estimate Dose Rates from Unpainted Concrete, EL 282' 6"
d) Figure 1 - Location of Concrete Types
e) Table 4 - Dose Rate Decontamination Factors
f) Table 5 - Comparison of Gamma Specification Data with Observed Dose Rates
g) Figure 2 - Location of Surveys EL 305'
h) Figure 3 - Location of Surveys EL 347' 6"
i) Figure 4 - Measured Permeability of Limestone Concrete

SUMMARY AND CONCLUSION

1. Concrete is a poor ion-exchange media for both cesium and strontium. The most credible mechanism for contamination intrusion

into unprotected concrete surfaces is mechanical absorption and retention of contaminated water. Utilizing the void fraction of the concrete microstructure and the physical water absorption of concrete blocks, it is possible to estimate the specific concentration of radionuclides in the concrete. (See Table 1.) Since all paints are, to some degree, water permeable, not even painted concrete surfaces are totally free from the possibility of internal contamination.

The depth and profile of this intrusion into the various types of concrete are not readily available. In the absence of any firm data, a uniform profile has been assumed. The depth of intrusion has been chosen for each type of concrete based upon expected physical conditions of the concrete. (See Table 2.)

These data can be used to estimate post-decontamination dose rates from these contaminated concrete structures. Hand calculations based on these values have resulted in contact dose rates of 1-64 R/hr on the 282' 6" elevation. (See Table 3.)

2. No conventional decontamination technique is universally effective on concrete surfaces. The results of the TMI-2 reactor building decontamination experiment (Ref. 1), the TMI-2 auxiliary building decontamination experience (Ref. 2) and the cleanup activities at the Phoebus 1A nuclear rocket motor test cell (Ref. 3) all indicate that conventional decontamination techniques are often ineffectual on concrete. However, there is direct evidence that both cesium and strontium leach readily out of some types of concrete (Refs. 2 and 4). This raises the possibility of removing internal contamination from concrete without gross removal of structural materials. In order to leach isotopes out of the concrete, it is necessary to keep the concrete surface wet with low concentration water for a period of days or weeks. Using hot water should enhance the leach rate due to the increased permeability of concrete at elevated temperatures. 2

3. Strontium and other airborne contamination evolving from the walls can be reduced by ventilation, by effective decontamination, and/or by fixing the contamination to the walls by painting.

RECOMMENDATIONS

1. The estimated source terms of this Technical Memorandum should be conservative and suitable for dose assessment estimates.

2. A leaching experiment should be performed to test the hypothesis that cesium and strontium can be easily removed from concrete. A potential leaching experiment is already underway in the basement where the water level is constantly being raised by leakage and decon activities, and lowered by pumping it through SDS. If no

leaching is taking place, the specific activity of the sump water should be decreasing by dilution. If careful measurements of radionuclide concentrations reveal that the specific activity is remaining constant, or is significantly higher than the predicted dilution levels, it should be indicative of leaching.

A detailed leaching experiment should also be considered for elevations 305' and 347' 6". This test should vary flow rates, temperatures and pH of leachate to determine the optimum leachate characteristics.

3. The concrete sampling by drilling in the reactor building was inconclusive. The only method to determine the extent of contamination in the concrete structures is to take core samples from some chosen locations and to perform a laboratory analysis on them. Depth of core bore need not exceed 1 or 2 inches and will not interfere with rebar location or the structural integrity of the reactor building. Leaching tests could also be done on core samples.

   A parallel effort should be undertaken to determine the contamination levels of the paint. Representative paint samples from various surfaces should be taken for analyses. If contamination levels are found, a complete paint evaluation should be undertaken by an independent testing company, such as K.T.A.-Tator, Inc. of Pittsburg, Pennsylvania, to determine paint film thickness, hardness and adhesion.

4. Airborne contamination levels may be reduced by a successful decontamination effort, improved ventilation air flow, or control envelopes for personnel. However, a more direct control of airborne releases from concrete surfaces can be effected by painting the surfaces to fix the contaminants.

5. All concrete surfaces in the reactor building basement should be maintained as wet as practical. This will decontaminate the concrete on an ongoing basis and will prevent dry out and the concomitant concentration of subsurface contamination. 2

DISCUSSION

Concrete Contamination Mechanisms

There are four major types of concrete surfaces inside the TMI-2 reactor building: (1) 5000 psi poured concrete, (2) 3000 psi poured concrete, (3) solid concrete blocks, and (4) hollow concrete blocks. The locations of these concrete structures in the basement are shown on Figure 1. All concrete structures on the upper elevations are made of 5000 psi concrete, with the exception of the Stair No. 2 and elevator enclosure (Ref. 5).

The 5000 psi concrete is Portland Type II cement with 3/4 in. limestone and washed sand aggregate. It has approximately 5% entrained air and a density of approximately 145 lbs./cu. ft. (Ref. 6). The 3000 psi concrete is of similar composition, but has a maximum density of only 115 lbs./cu. ft.

There are three mechanisms which can determine the amount of contamination penetrating into unpainted concrete structures exposed to contaminated water: intrusion, ion exchange, and diffusion. Intrusion is the mechanical penetration of contaminated water into the porous microstructure of concrete. Ion exchange is the chemical attachment of certain isotopes to specific concrete fractions. Diffusion is the transfer of isotopes from regions of high concentrations to regions of lower concentrations within the concrete. These three mechanisms work together to transport contamination into concrete.

Under normal conditions, water in contact with unpainted concrete will penetrate into the concrete by filling the macroscopic and microscopic voids remaining in the concrete. The permissible void fraction of 5000 psi concrete is 4-6%. The ultimate depth of penetration is a function of time, and the physical properties of the concrete and the water. If ion exchange processes take place between isotopes in the water and specific concrete fractions, the concentration of the affected isotopes in the water inside the concrete will decrease as the water penetrates. As soon as a concentration gradient exists in the water, diffusion of the isotopes in question will begin to carry more isotopes into the concrete where it will ultimately be tied up on ion exchange sites or maintain equilibrium with the sump water. It should be noted that painted concrete will react in much the same manner if the period of contact is long. All polymer films (paints) are water permeable to some extent. The effect of paint would be slow to the rate of concrete intrusion.

Very little data is available in the literature to describe the absorption of contamination into concrete. However, an experiment conducted by Brookhaven National Laboratories indicates that concrete is a poor ion exchange media for cesium (Ref. 4). This experiment was conducted to ascertain the leaching characteristics of various radwaste solidification media. Like concentrations of three common isotopes (Cs 137, Sr 85, and Co 60) were matrixed in a Portland Type II cement sample with a water to concrete ratio of 1.0. These samples were allowed to cure for three days. At the end of that time, the amount of Cs 137 in the free standing water was measured and found to be roughly equal to the amount in the initial sample. Hence concrete poses little more than a physical barrier to cesium migration.

The other isotopes in this experiment were found to have weak ion exchange characteristics with Portland Type II cement. The amount of strontium in the free standing water was found to be roughly an order

of magnitude lower than that of the original sample. The amount of Cobalt was found to be roughly two orders of magnitude lower than that of the original sample. "This behavior is indicative of an ion exchange process occurring within cement where ionic selectivity generally increases as the valence increased and as the ionic radii decreases for a given valence" (Ref. 4).

In order to assess the potential significance of absorbed contamination in the TMI-2 reactor building concrete, estimates were made for the dose rates in the basement. Since cesium is the major gamma ray emitting isotope of concern at TMI-2, and since concrete is a very poor ion exchange medium for cesium, the major contamination mechanism for concrete structures in the Reactor Building is mechanical retention of water containing cesium in the void spaces of the concrete. According to the American Concrete Institute (Ref. 7), the void space allowed in concrete with 3/4" coarse aggregate is 4-8%. Hence, the source term for exposed concrete surfaces can be represented by assuming 5% of the concrete volume is filled with water containing a representative specific concentration of cesium isotopes.

Concrete block is much more porous than poured concrete. According to ANSI/ASTM Standard C90-75 (Ref. 8), "Normal Weight" general use (Grade N) concrete block material can absorb a maximum of 13 lbs. of water/cubic foot (21% by volume). This value applies to both hollow and solid concrete block material. A single standard (8"x8"x16" nominal) solid block contains approximately .593 ft^3 of concrete. A hollow block contains approximately .270 ft^3 of concrete with the remaining .323 ft^3 being the hollow spaces which, in the case of the enclosed stairway (Stair No. 2) walls, can be assumed to be filled with sump water (although it is expected that this water will seep out now that the building water level has been lowered).

Cesium Contaminated Concrete Source Terms

Utilizing this model for the contamination of the unpainted concrete surfaces, and the specific concentrations of the two cesium isotopes in the sump water contained in Ref. 9, the specific concentrations of cesium 134 and 137 can be easily calculated. (See Table 1.)

It must be noted that these specific concentrations alone are insufficient for dose assessment analyses. Some estimate of the depth and the profile of the sump water intrusion into the concrete surfaces must be made. Intuitively, the depth of intrusion will decrease and the profile will be more differentiated as the concrete density increases. Poured concrete is the densest of these materials and is relatively impervious to water, however microcracking under tension, as well as surface imperfections, would provide some pathways for water intrusion. Concrete blocks have a much more open microstructure which would indicate that water could credibly intrude deeply into the

material. This is especially true of hollow concrete blocks, which would be completely water logged after 2 years of immersion.

In the auxiliary building, it was determined through core boring data that contamination had penetrated untreated, non-troweled concrete surfaces to a depth of .8 inches, and untreated, trowel finished concrete surfaces to a depth of .25 inches (Ref. 2). These penetration depths were the result of an immersion time less than a few weeks. Therefore, they should represent a minimum expected value for areas submerged or kept wet for long periods of time in the reactor building. The only method to determine the actual depths of penetration and profile of intrusion in the reactor building is to take core samples of the concrete. 2

This was attempted by EG&G but the procedure employed seems to have yielded inconclusive results (Refs. 1 and 14). The procedure, which consisted of milling several holes in the surface to different depths with a small drill bit while vacuuming the dust, seems to have picked up only variations in the surface contamination levels. Unless a method exists to interpret the drilled sample data, it appears that the only way to proceed is to take new core samples of the concrete for laboratory analysis. These core samples need not be deeper than one or two inches in depth in poured concrete on the 305' or 347' 6" elevations due to the low permeability of dense structural concrete. Hence, the rebar need not be exposed and no permanent structural losses will occur. Conversations with personnel at Brookhaven National Laboratories (Refs. 10, 11 and 12) have confirmed the difficulty of analytically predicting the depth and profile of water intrusion into concrete without core bore data.

Dose Rates from Concrete

In the absence of data, the concentrations of cesium in the concrete must be assumed to be uniform throughout the depth of intrusion. The assumed depth of intrusion for each type of concrete is summarized in Table 2. A parametric study of the effect of intrusion depth on dose rates indicates that 90% of the dose rate comes from the outer 6 inches of concrete. Therefore, water intrusion into concrete beyond this depth is not a significant factor with respect to dose rates.

Dose rates from the concrete surfaces in the TMI-2 Reactor Building basement have been estimated utilizing the assumptions in this memorandum and are summarized on Table 3. These dose rates are based on hand calculations using approximate source geometries to represent the concrete surfaces (Ref. 13). The dose rates determined for Stair No. 2 (i.e., the hollow concrete block) can be compared to the actual dose rate measured in Stair No. 2 of 22 R/hr.

For the 305' and the 347' 6" floors, actual survey data has been collected by three different groups: the decontamination experiment 2

performed by Bechtel (Ref. 1), the gamma-ray spectrometer scans performed by Science Applications, Inc. (Ref. 1, Appendix I), and a range of experiments performed by EG&G (Ref. 14). These independent surveys took place over a period spanning the decontamination experiment and indicate that the efforts to decontaminate concrete surfaces did not succeed in reducing dose rates significantly. In fact, the SAI gamma spectrometer scans seem to indicate that the surface contamination actually increased after decontamination (see Table 4).

The EG&G experiment generated decontamination factors (DFs) an order of magnitude lower than reported by Bechtel in Reference 1. This was due to the fact that Bechtel computed DFs based on swipe data rather than on dose rate decreases. This was done to determine the effectiveness of the decontamination efforts in reducing smearable surface contamination. However, these DF's are not applicable to a dose rate reduction projection. Relying on swipe data to compute DFs assumes that all contamination is on the surface. The TMI-2 auxiliary building experience and the conclusions of the Bechtel report refute such an assumption. Bechtel contact dose rate data (Ref. 1, Appendix C) results in DFs consistent with the DFs generated from the EG&G data (see Table 4) which indicates the presence of subsurface or tightly bound contamination.

Using dose rates to generate DFs assumes that the contamination is the major source in the area. To check this assumption, the SAI gamma spectroscope data was used in a simple analytical model to develop both area and contact dose rates. The model developed consisted of a semi-infinite source shaped like the intersection of a floor and a straight wall, and considered only the dose contribution from ten feet wide sections of surface (infinitely long). Area dose points were 3 feet from each surface. Contact dose rates were 3 feet from one and 1 inch from the contact surface. Where actual floor and wall scans for an area were performed, the concentrations were modeled exactly. Where only one surface scan was performed, it was assumed that the wall and floor had the same concentration. The results of this calculation (Ref. 16) are presented in Table 5 along with actual survey data taken by Bechtel (Ref. 1).

The calculated dose rates are consistently the same order of magnitude as, and often within a few percent of, the observed data. Much of the apparent difference may be explained by random variations in the source strength and by the fact that the exact locations of the survey points in the two reports were often difficult to correlate. In addition, at the time of these measurements, the dose rates through the concrete floor on the 305' level from the sump water in the basement were probably in the 100-200 mR/hr range (Ref. 15). In order to eliminate this contribution, SAI aligned their collimated gamma ray spectrometer at a 45° angle to the floor. This procedure blocked dose contributions from the basement that Bechtel's survey would detect. Also, if the concrete floor was contaminated internally, this proce-

dure would further reduce the apparent dose rate since the increased self absorption due to the slant angle will predominate the larger surface area viewed by the detector. Hence, it might be expected that both area and contact dose rates predicted by the SAI data on the 305' elevation would be lower than the actual dose rates measured by Bechtel. 2

In short, the measured dose rates generally agree with the measured levels of contamination apparent on the walls. This implies that the general area dose rates are a result of the contaminated walls and floors. The Bechtel data seems to indicate that the decontamination experiment resulted in a significant reduction of smearable contamination. The failure to obtain a similar reduction of dose rates indicates that the smearable contamination was not a significant contribution to the total dose rates and that the dose rates result from contamination that is either absorbed by the concrete, the paint, or both.

Leaching Contamination out of Concrete

Contamination absorbed into the concrete will not be removed by conventional decontamination techniques. An indication of how difficult decontaminating concrete structures can be is given in the discussion of the cleanup activities undertaken at the Phoebus 1A test cell after a reactor accident (Ref. 3). After that accident, "(a) variety of techniques (was) used to remove contamination from the concrete, including washing with high pressure hoses and water; scrubbing with street brooms and mops; wet vacuum cleaning; dry vacuum cleaning; and scrubbing with Versine, Oakite and four different Turco solutions. Sweeping compound was also used in some areas. It was notable that some agents were effective on particular areas of concrete, but not on other areas. No one agent was universally effective, and none reduced contamination levels to acceptable values on the reactor pad, the new dewar pad, the test cell roof, or the area just behind the test cell building."

At TMI-2, Vikem had similar difficulties getting concrete surfaces clean in the Auxiliary and Fuel Handling Buildings (Ref. 2). Repeated decontamination of surfaces was required to reduce a contamination to acceptable levels. Often, apparent recontamination occurred as a result of the movement of subsurface contamination to the surface. Repeated scrubbing, wet vacuuming and "sweating" of concrete was utilized with variable success. Special leachout chemicals were prohibited due to the potential sensitivity of the Auxiliary Building Cleanup resins to these agents. 2

If the contamination is trapped in the paint, decontamination can be facilitated by removing the existing paint and repainting the exposed surfaces. If the contamination is in the concrete, it must be removed by gross removal of concrete or by selective leaching of the

isotopes out of the concrete. Clearly, leaching the radionuclides out of the concrete is less expensive, and less damaging to the reactor building structures.

The primary mechanism involved in the "leaching" process is the evolution of water out of the concrete. Contaminated liquid which has intruded deeply into a flooded concrete structure begins to move toward the surface(s) as soon as the flood water level is lowered. Water trapped on, or very near, the outer surface, essentially flows off of the concrete. Water trapped more deeply wicks to the surface and evaporates, leaving behind its contamination. This explains the apparent recontamination of cleaned areas in the auxiliary building (Ref. 2). As the concrete dries, the area where this evaporation takes place moves deeper into the concrete, thus concentrating contaminants inside the concrete itself. This explains why it is often necessary to remove the outer surface of a concrete structure in order to affect final decontamination.

This drying process can be minimized by a process known as "sweating" the concrete. (Ref. 2) In this process, a herculite sheet is taped over a concrete surface after initial decontamination. The herculite prevents evaporation from the surface so that the water from inside the concrete can flow to the surface and can thereby transport the contamination out of the concrete. The water collects on the herculite where it can be drained or mopped up conventionally.

This sweating technique will not be as effective if the concrete has dried out in an uncontrolled manner. The contaminants will already be trapped in concrete and it is unlikely that the flow of water vapor will be adequate to transport them to the surface. A Supervac or chemical leachste, such as Nutech 700, can be used to remove some of this subsurface contamination. However, due to the transient nature of these expediants, they are not able to remove all subsurface contamination. 2

A true leaching effect could be encouraged by keeping contaminated concrete surfaces wet with non-contaminated water. This water will intrude into the concrete and re-dissolve the radionuclide trapped in the concrete. Once this water is contaminated, a concentration gradient will exist between it and the water on the surface of the concrete. This will cause the radionuclides to diffuse to the surface where they will be washed away by the flow of water. The flow of the water need not be great (.1 - 1.0 gpm/100 ft.^{2}) and need not interfere with ongoing recovery efforts. The water would not be highly contaminated due to the slowness of the diffusion process.

The time required for such a leaching process to decontaminate concrete will be quite long. Indeed, such a process may only serve to reduce the dose rates and to prevent airborne evolution from the concrete until more conventional techniques can be employed. The ul-

timate time required for a leaching-only approach is a function of the permeability of the concrete, the depth of the contamination, or the solubility of the contaminating isotopes in the leachout. The only factor that seems to be a realistic variable for application at TMI-2 is the permeability of the concrete. 2

According to experiments done by the Hanford Engineering Development Laboratory (Ref. 17), the permeability of limestone concrete similar to TMI-2 concrete is a strong function of temperature (see Figure 4). The extremely strong variation in permeability with relatively low temperatures (i.e., less than 200°F) suggests that large increases in the leachout rate are possible with small elevations in temperature.

Since cesium is very soluble in water without chemical adjustments, it does not seem that extraordinary measures to increase the solubility will result in large reductions in decontamination time. 2

TABLE 1

Cesium Source Terms for Unpainted Concrete EL 282' 6"

Source Description	Maximum Specific Concentrations (μc/cc) Cs 134	Cs 137
Poured Concrete	1.25	8
Solid Block	5.22	33.38
Hollow Block (Hollows Empty)*	2.38	15.20
Hollow Block (Hollows Flooded)*	15.99	102.35
Sump Water (Reference 5)**	25	160

Notes: * These assume a homogeneous source material and water density of 62.305 lbm/ft^3.

** Water concentrations as of July 1980 (Ref. 9) 160 μCi/ml of Cs 137 and 25 μCi/ml of Cs 134.

TABLE 2

Water Intrusion in TMI-2 Reactor Building Concrete

Concrete Surface	Depth of Intrusion	Profile
Poured	.8"	Uniform
Solid Block	18"	Uniform
Hollow Block	18"	Uniform

TABLE 3

Estimated Dose Rates from Unpainted Concrete EL 282' 6"

Source	Contact Dose Rates (1" from Surface)
Poured Concrete: 5000#	1.34 R/hr
3000#	1.37 R/hr
Concrete Blocks:	
Solid	22 R/hr
Hollow Flooded	64 R/hr
Hollow Unflooded	12.5 R/hr

Note: Intrusion of Cs 137 and Cs 134 from water containing 160 μCi/ml of Cs 137 and 25 μCi/ml of Cs 134.

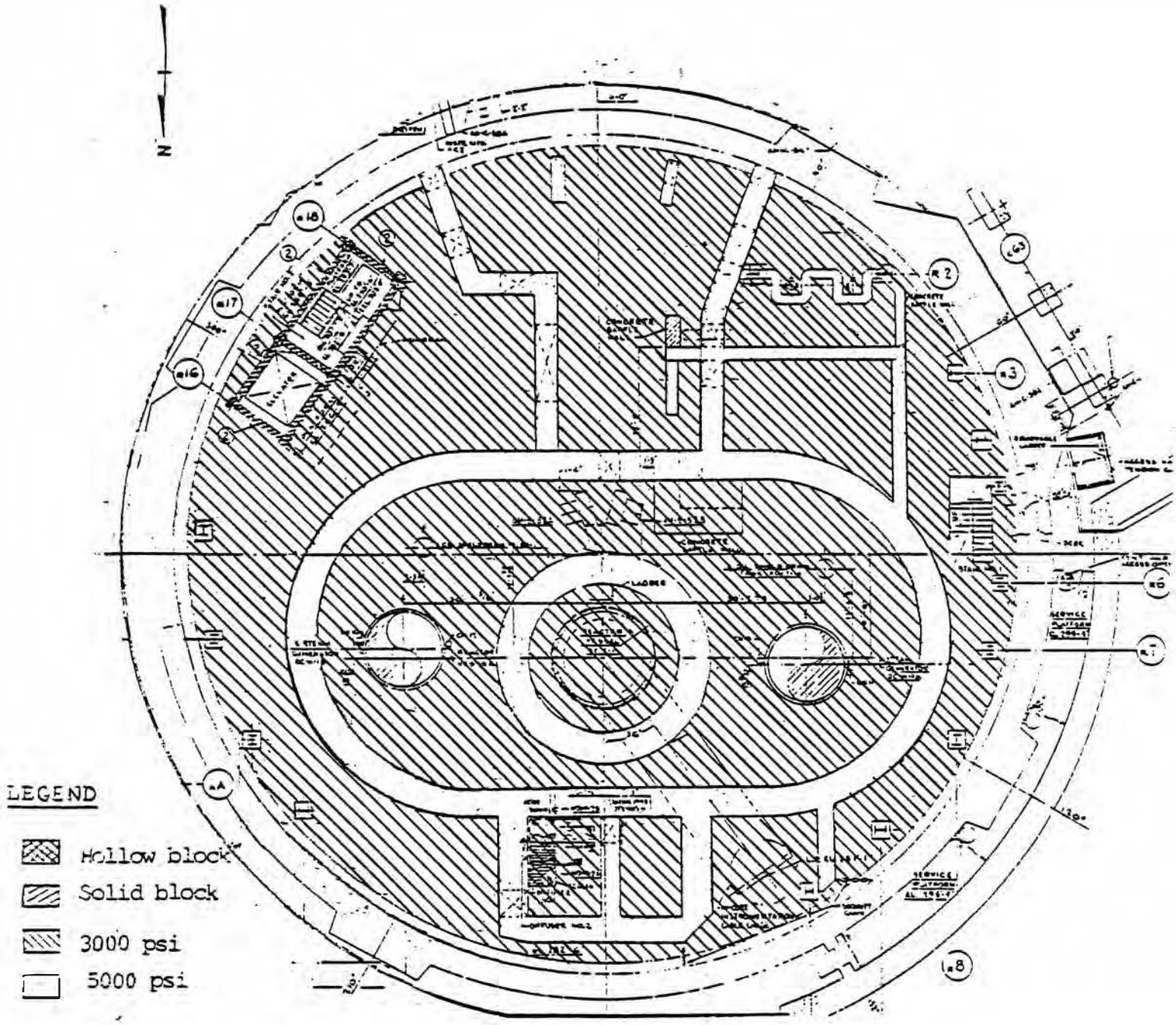


Figure 1 - Location of Concrete Types

TABLE 4 - DOSE RATE DECONTAMINATION FACTORS

FCAG RU-2 DATA

ELEVATION	DATA POINT	VERTICAL OR HORIZONTAL	PRE-DECON α mR/hr	POST DECON α mR/hr	DF	PRE-DECON β mR/hr	POST DECON β mR/hr	DF
305'	08-2	V	1000	300	3.3	~0	400	0
	11W	V	900	300	3.0	~0	~0	-
	07	H	250	80	3.1	560	050	11.2
	06	H	500	240	2.0	7000	500	11.8
	05B	H	400	150	2.7	5500	~0	∞
	03	H	1000	1000	1	5700	~0	∞
	02	H	1500	500	3.0	19000	9900	0.5
	67W	V	200	600	0.5	37	~0	∞
	70	H	160	500	0.3	2000	~0	∞
347'-6"	51W	V	35	16	1.0	70	48	1.5
	51	H	200	60	3.3	700	312	2.2
	49	H	300	160	1.9	1400	468	3.0
	36W	V	65	50	1.3	35	40	0.9
	04	V	75	80	0.9	70	80	0.9
	16W	V	75	50	1.5	35	40	0.9
	02-2	V	120	100	1.2	525	400	1.3
	44	H	500	180	2.8	7550	468	16.1
	0004	H	300	150	2.0	3150	1287	2.4
	02	V	110	70	1.6	35	40	0.9
	50W	V	165	3000	1.7	210	80	2.6
	1434	H	750	450	1.7	4300	975	4.5
	0101	H	500	220	2.3	6150	1872	3.3
	01-2	V	150	100	1.5	1105	160	0.7
	36W	V	65	16	1.3	123	96	1.1
367'	2210	V	325	260	1.3	158	244	0.6

AVERAGE DF VALUES

305'	HORIZONTAL	α - 2.00	β - 5.10
	VERTICAL	2.27	-
347'-6"	HORIZONTAL	2.13	5.25
	VERTICAL	1.33	1.2
367'	VERTICAL	1.3	0.6

BECHTEL GROSS DECON EXPERIMENT

ELEVATION	DATA POINT	VERTICAL OR HORIZONTAL	PRE-DECON α mR/hr	POST DECON α mR/hr	DF	PRE-DECON β mR/hr	POST DECON β mR/hr	DF
305'	12-1	H	150	190	0.2	6100	1900	3.2
	12-2	H	150	1200	0.3	722	104	2.4
	12-3	H	150	120	0.3	494	114	4.3
	12-4	H	160	150	0.9	190	76	2.5
	12-5	H	140	110	1.3	684	266	2.6
	1	H	290	250	1.2	1930	1330	1.5
	21	H	250	240	1.0	6100	914	5.3
	25	H	420	300	1.1	684	456	1.5
	29	H	1900	1400	1.4	9000	4290	2.1
	54	H	200	200	1	1590	507	3.1
	50	H	340	300	1.1	2320	1170	1.8
	53-3	H	900	1000	0.5	2660	700	3.4
347'-6"	29-1	H	160	140	1.1	1760	333	5.3
	29-2	H	100	60	1.7	480	105	2.6
	29-3	H	360	100	1.6	1760	296	5.9
	29-4	H	160	100	1.6	2560	222	11.5
	29-5	H	180	100	1.8	2480	148	16.8
	87	H	260	180	1.4	4560	2294	2.0
	89-1	H	160	110	1.5	1760	701	2.5
	89-2	H	120	120	1	2720	666	4.1
	89-3	H	160	80	2	2560	777	3.3
	89-4	H	240	130	1.8	7040	1147	6.1
	89-5	H	120	60	2	1920	333	5.8
	91-1	H	50	40	1.3	440	333	1.3
	91-2	H	60	40	1.3	400	341	1.4
	91-3	H	80	60	1.3	560	407	1.4
	91-4	H	110	70	1.1	720	333	2.2
	91-5	H	180	130	1.4	2480	1250	2.0
	112	H	160	90	1.8	2496	460	5.3
	62-1	H	140	100	1.4	850	741	1.2
	62-2	H	140	90	1.3	468	468	1
	62-3	H	110	80	1.4	661	234	2.8
	62-4	H	110	80	1.4	5115	3901	1.5
	62-5	H	100	150	1.2	1690	1170	1.4
	86	H	220	180	1.2	3042	20201	1.5
	1150	H	120	80	1.5	202	468	1.5

AVERAGE DF VALUES

305'	HORIZONTAL	α - 1.11	β - 5.25
347'-6"	HORIZONTAL	1.46	1.77

TABLE 5

COMPARISON OF GAMMA SPEC DATA WITH OBSERVED DOSE RATES

Area Designation			Apparent Contamination $\mu Ci/cm^2$ 1)		Estimated Area Dose Rates (mR/hr) 2)		Observed Area Dose Rates (mR/hr) 3)		Estimated Contact Dose Rates (mR/hr) 2)		Observed Contact Dose Rates (mR/hr) 3)	
	SAI	Bechtel	Pre-Decon	Post-Decon	Pre-Decon	Post-Decon	Pre-Decon	Post-Decon	Pre-Decon	Post-Decon	Pre-Decon	Post-Decon
305-1	13	-	<.2	.5 ± .2	37	29	-	-	48	71	-	-
305-2	H7	12*	.96 ± .10	.4 ± .2	37	29	170	120	120	61	290	130
305-3	V9	-	.5 ± .3	<.6	32	38	-	-	74	88	-	-
305-4	34	34	.9 ± .2	5.3 ± .4	58	340	400	330	130	780	340	280
305-5	H5	-	<.3	-	19	-	-	-	44	-	-	-
347-1	33	33*	<.14	1.1 ± .2	9	70	90	70	20	160	120	100
347-2	H10	39*	.5 ± .16	.8 ± .2	32	51	100	90	74	120	200	150
347-3	149	48*	.97 ± .17	2.0 ± .1	62	130	140	110	140	290	270	150
347-4	H9	54*	.85 ± .17	-	54	-	180	150	130	-	330	170
347-5	36	-	<.20	<.30	37	19	-	-	48	44	-	-
347-6	36-A	35*	.96 ± .15	-	37	-	150	120	120	-	270	210

* Five measurements have been averaged to yield the Ref. 10 Dose Rates.

1) Data from Ref. 1, Table 4, Appendix I
2) Calculated Dose Rate from Ref. 12
3) Data from Ref. 1

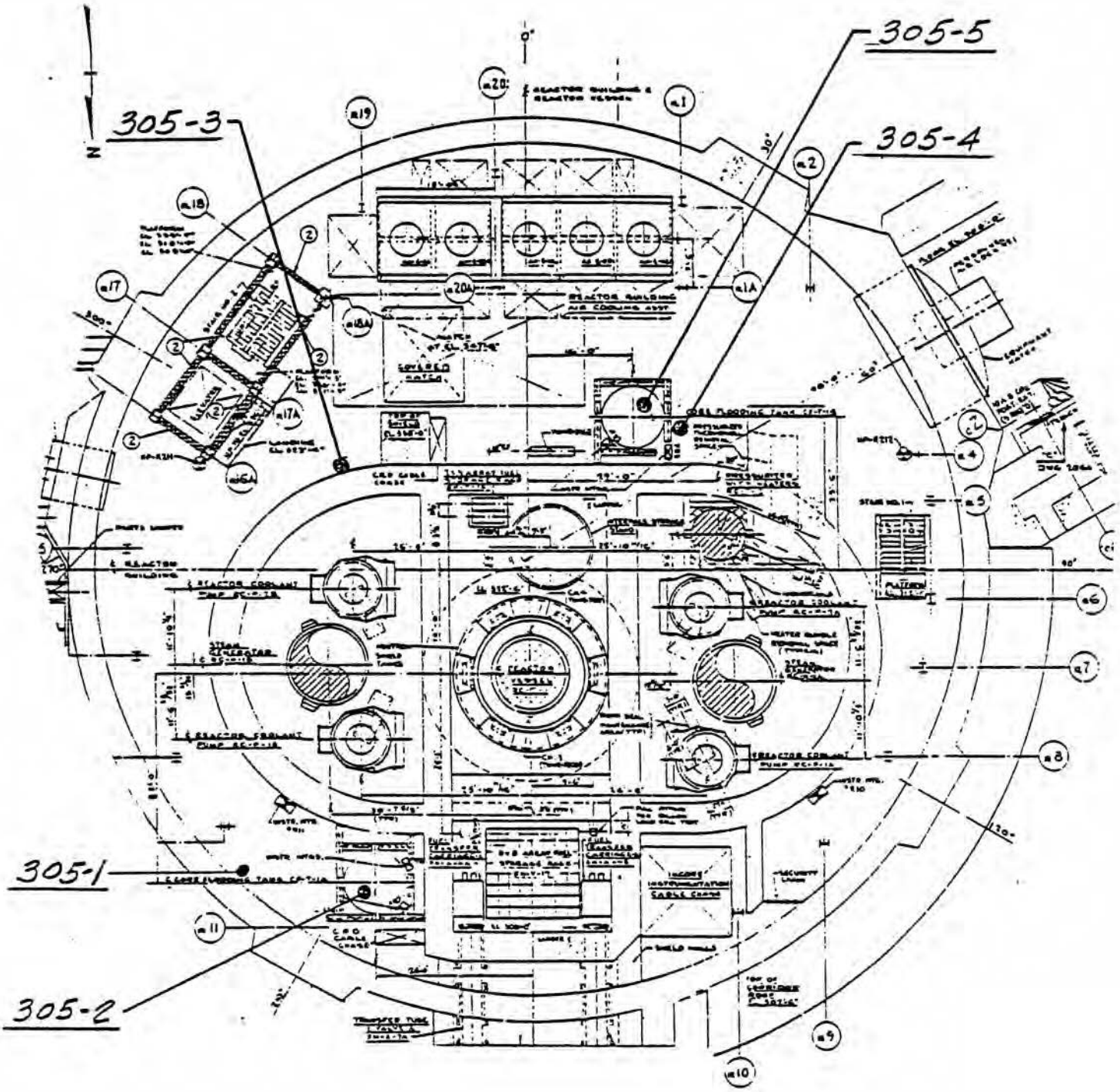


Figure 2 - Location of Surveys 305

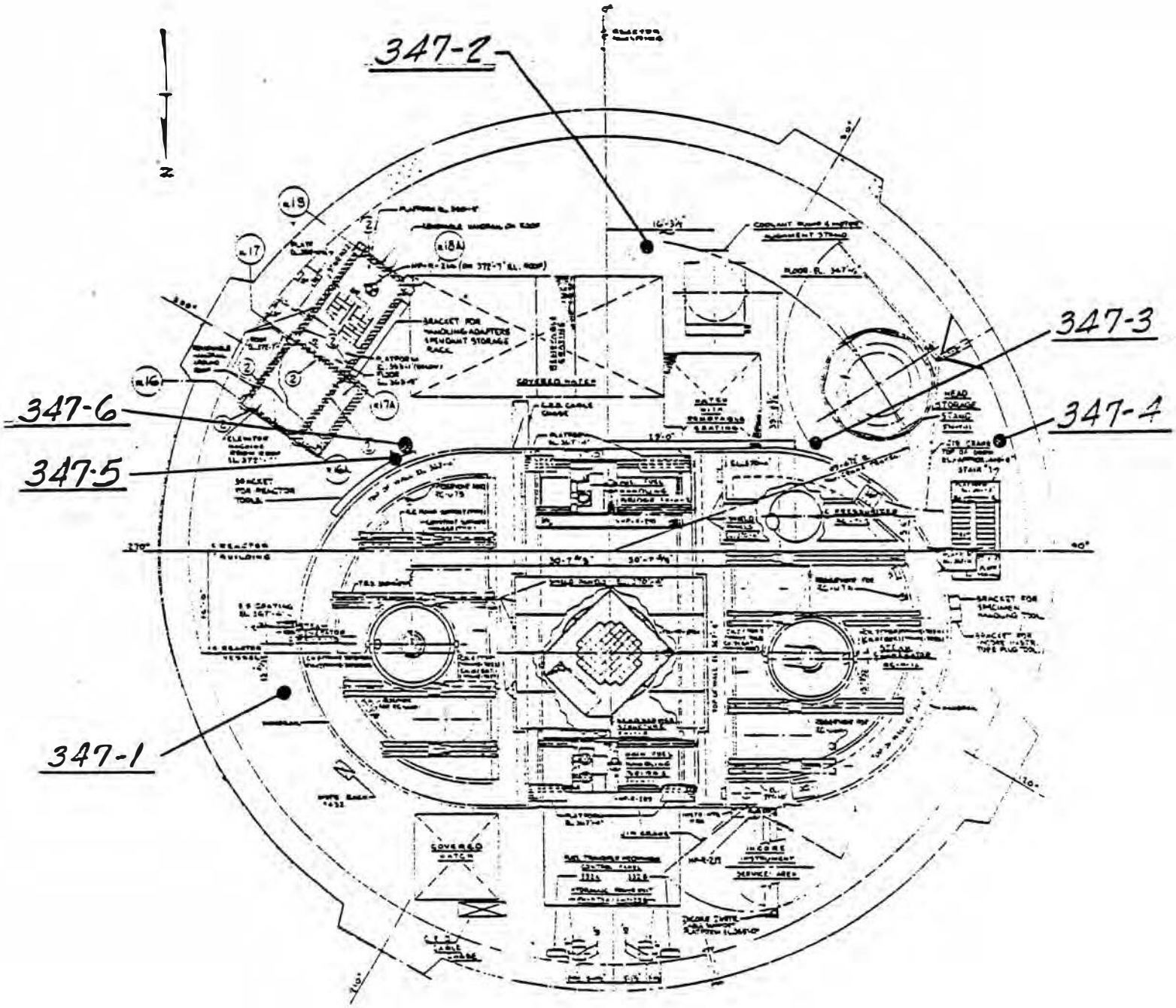


Figure 3 - Location of Surveys 347'6"

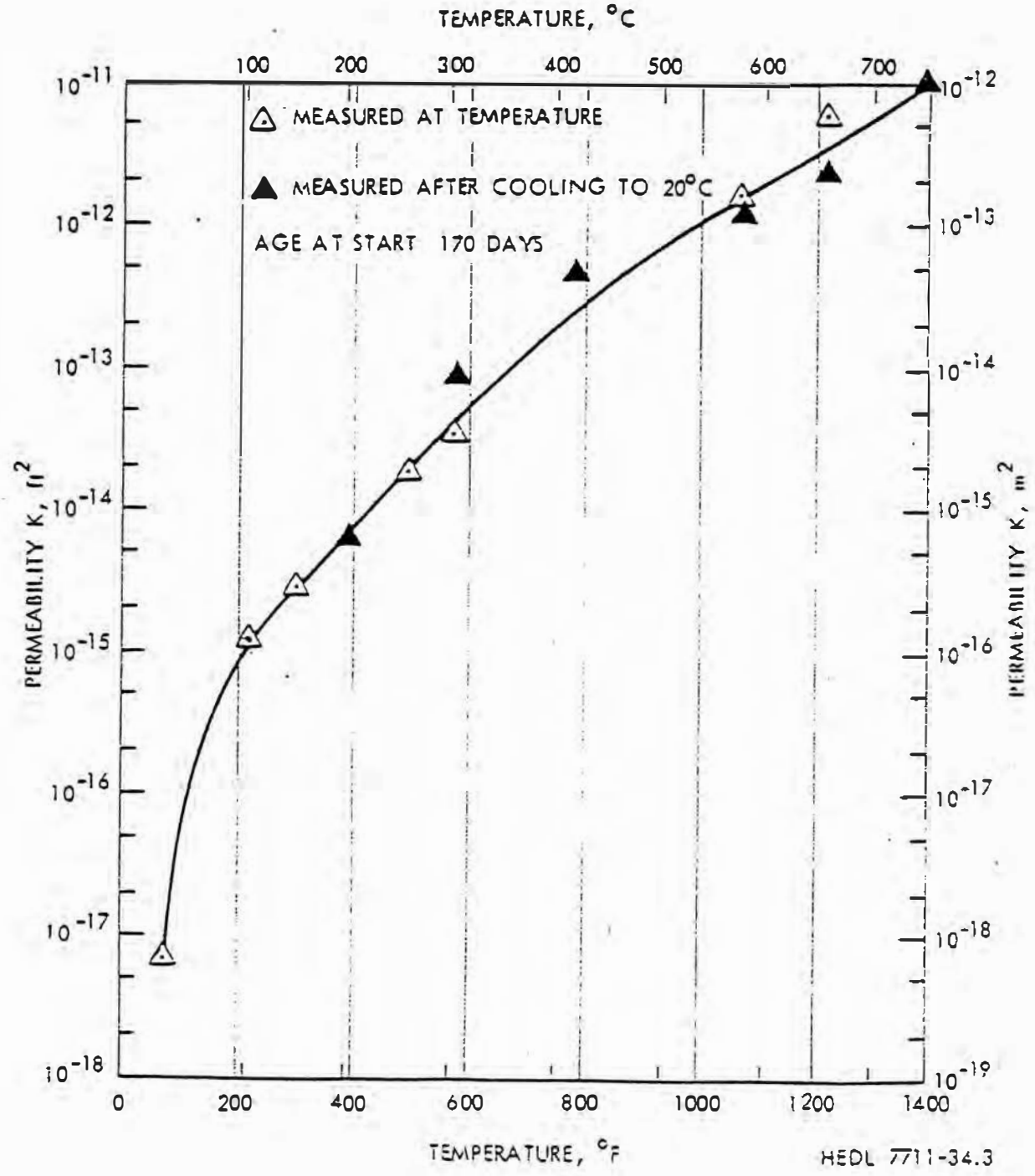


Figure 4 - Measured Permeability of Limestone Concrete.

ATTACHMENT IV-3

TECHNICAL MEMORANDUM

DATE 12/9/82

NUMBER
TM 3680-7 Rev. 0

COPIES TO:
Oradell Library
Woodbury Library
Jacksonville Library
WWhite
ASDam-db
FJPatti
NLacy
RGagliardo
JMTuohy
RBurrow
VFricke
CWHess
pf 2 3680-01

TO E. J. WAGNER

FROM S. S. ROSEN

SUBJECT W.O. 3680-01
TECHNICAL ASSISTANCE AND ADVISORY GROUP
TMI-2 RECOVERY
LEACH RATE OF ACTIVITY INTO REACTOR BUILDING SUMP WATER

KEYWORDS TMI-2, REACTORS, DECONTAMINATION, LEACHING, CONCRETE

REFERENCE:

(1) ORNL/TM 7448 DATED JULY 1980

(2) GEND-INF-011 "CONTAINMENT SUMP RADIONUCLIDE DISTRIBUTION STUDIES" T.E. COX, et al, EG&G, 8/82

(3) PRIVATE COMMUNICATION MS. C. HITZ, GPU TO S. S. ROSEN, B&R TRANSMITTING ORNL AND GPU RADIOCHEMICAL ANALYSES OF SUMP WATER

(4) ORNL MEMO, J.A. CARTER TO D.O. CAMPBELL "TMI-II SLUDGE DATA" DATED 10/27/82

(5) PRIVATE COMMUNICATION, TOM LOOKABILL, NUS TO S.S. ROSEN

(6) B&R CALCULATION NO. 3680-15-15.

(7) B&R CALCULATION NO. 3680-15-14.

(8) B&R CALCULATION NO. 3680-15-16.

(9) B&R TECH. MEMO NO. 3680-3 REV. 2, "CONTAMINATION INTRUSION INTO CONCRETE," 11/17/82.

(10) TABLE 1 ESTIMATED CURIES IN CONTAMINATED STRUCTURES

PURPOSE

High gamma dose rates are continuing to be experienced between the 282-6 and 305 elevations in the TMI-2 reactor building after draining the sump water. Suspected sources include Cs-137 in the sludge on the 282-6 floor and intruded into the long-flooded concrete. This evaluation is performed to attempt to determine the source based upon the appearance of Cs-137 and Sr-90 in the sump water as it is diluted and pumped out of the reactor building. The evaluation is intended to assist in dose rate reduction and decontamination planning.

CONCLUSIONS

1. During dilution of the reactor building sump water occurring during decontamination activities, Cs-137 and Sr-90 isotope balances indicate that a reservoir of activity is adding soluble nuclides to the sump water.

2. Evaluation of the sludge sample taken on 6/24/82 indicates that the sludge layer on the floor of the 282'-6" El. contained a reservoir of 3672 Ci of Cs-137 and 1010 Ci of Sr-90.

3. From the data available it is not clear whether the activity appearing in the sump water is from the sludge layer or from sources within the concrete of the 282'-6" El. or from another source.

4. A proposed analytical model to explain the transport of activity from the sludge/concrete to the sump water was developed and application of available data indicates a relatively long time constant for the activity transfer between the reservoir to the sump water of about 127 days for Cs-137. Data was not available to develop a similar constant for Sr-90.

RECOMMENDATIONS

1. In order to develop a decontamination action plan for the 282'-6" El. that results in procedures to minimize exposures (ALARA), the source of the reservoir of contamination appearing in the sump water should be determined. If it is only the sludge layer, then removal of the activity from the sludge would have the highest priority, but if it is within the concrete, then procedures to enhance the leaching of that activity out of the concrete should be instituted. Experiments to develop data to enable enhanced leaching rates would then need to be designed including the possibility of taking core borings of the concrete.

2. Decontamination activities within the reactor building are continuing, diluting the sump water further. The nuclide concentrations in the sump water should be determined during the next pumpout to further define the activity transport from the reservoir.

3. Additional samples of the sludge from the 282'-6" El. should be obtained and analyzed to observe its depletion as an activity source and thereby to determine if the concrete structure as well as the sludge layer is a reservoir for activity.

4. The sludge layer should be hosed down with warm water to decrease its activity release time constant and help transport its held activity into the sump water so it can be pumped out of the reactor building to decrease the dose rates.

5. Washing of building surfaces and consequent dilution of the sump water should continue as long as it is effective in reducing activity concentrations and dose rates. Activity concentrations in the sump water and sludge should continue to be monitored as well as dose rates using TLD trees.

BACKGROUND

In July of 1980 about 529,000 gallons of water were in the TMI-2 reactor building sump with a Cs-137 concentration of 160 uCi/ml[1]. From July of 1980 until September of 1981 no water was removed from the reactor building sump, and water was added from continuing primary coolant system leakage to bring the total volume in the sump to about 606,000 gallons with a measured Cs-137 activity of 137 uCi/ml. This Cs-137 concentration is consistent considering that an additional 7400 Ci of Cs-137 were introduced into the containment sump with the reactor system leakage over that same time interval. Between September of 1981 and February of 1982 about 601,000 gallons of water in 16 stages (batches) were pumped out of the containment sump with some continued input to the sump from reactor system leakage of about 0.1 GPM. During this period the Cs-137 activity remained fairly constant at about 130 uCi/ml[3] indicating an equilibrium situation. During this same period the Sr-90 activity remained also relatively constant at about 5.4 uCi/ml[3] also indicating an equilibrium situation for this isotope. (All these activity concentrations utilize the ORNL analyses which reference (3) advised were the more accurate.)

DILUTION AND CURIE BALANCE TO APRIL 1982

During March and April of 1982 about 13,000 gallons of water were added to the containment sump diluting the Cs-137 activity to 118 uCi/ml. This water was principally from decontamination activities. However, an activity balance comparing total Curies of Cs-137 before and after the dilution indicates that about 3300 Ci of Cs-137 appeared in the sump water, i.e. the diluted concentration should have been 97 uCi/ml of Cs-137 rather than the measured value of 118 uCi/ml. This indicates that a source of Cs-137 either from the sludge on the floor of the 282' El. or from within the concrete was solubilized and entered the liquid phase. This same phenomena occurred with the Sr-90 activity concentration in the sump water as it actually increased slightly to a measured value of 5.82 uCi/ml after the dilution.

SLUDGE SAMPLE EVALUATION

On June 24, 1982, sludge samples were taken from the floor of the 282' El. and sent to ORNL for analysis. The results of this analysis indicated the following[4]:

- For both Cs-137 and Cs-134 the supernate contained 53% of the total sample activity for those isotopes and an

additional 38% of the activity was easily washed from the solid portion leaving about 9% in the solids portion as insoluble.

- For Sr-90 the supernate contained 9% of the Sr-90 activity and the insoluble portion in the solids was 91% of the total.

Based on the ORNL sludge analysis and the observation that the sludge layer was about 1/2 to 3/4"[5] deep at the time of the sampling and assuming that the sludge layer was uniform over the entire floor area, the total Curies of insoluble Cs-137 and Sr-90 in the sludge layer are 312 and 924 respectively. These are a maximum values based on a 3/4" layer and considering the sludge to be of the same composition as that in the sample sent to ORNL for analysis[6]. In addition there is a relatively easily leachable Cs-137 component in the sludge containing 1414 Curies and 1946 Ci dissolved in the supernate for a total of 3672 Ci of Cs-137 in the sludge layer. Similarly there is a total of 1010 Ci of soluble plus insoluble Sr-90 in the sludge layer. The supernate portion of the sample contained 150 uCi/ml of Cs-137 whereas a sample of the sump water taken just a few days earlier on June 15, 1982, during sump pumpout No. 18 showed only 87 uCi/ml[3]. (This is a GPU measured value as an ORNL value was not available for this pumpout.) This indicates that the sump water was not in equilibrium with the sludge layer at that time. The same is true for the Sr-90 where the sludge sample supernate contained 6.93 uCi/ml and the sump water 5.4 uCi/ml.

DILUTION AND CURIE BALANCE SINCE APRIL 1982

Between the end of June 1982 and the end of September 1982 an additional 58,300 gallons of water was added to the sump at which time pumpout No. 19 was performed. The measured Cs-137 activity of the sump water was 21 uCi/ml[3]. A Curie balance for Cs-137 dissolved in the sump water between the 6/15/82 pumpout (No. 18) and the 9/28/82 pumpout (No. 19) indicates that about 390 Ci have appeared in solution[6]. Although the Cs-137 concentration decreased from 87 to 21 uCi/ml between pumpouts 18 and 19 due to dilution, the Sr-90 concentration actually increased from 5.3 uCi/ml to 5.6 uCi/ml indicating a large source reservoir of Sr-90. Note the sludge layer was estimated to contain 1010 Curies on 6/24/82. A Curie balance for Sr-90 between pumpouts 18 and 19 indicates that about 485 Curies have appeared in solution[6].

SOURCES OF Cs-137 AND Sr-90

From the above data and analyses it is not possible to determine whether the additional Cs and Sr appearing in solution are being solubilized from the sludge layer or from within the concrete at the 282' El. It is reasonable to assume that the mobility of

soluble solids is greater from the sludge than from the concrete. If it is assumed that all of the appeared Curies came from the sludge layer, then the sludge is being depleted and after pumpout No. 19 contains approximately 3283 Curies of Cs-137 and 525 Curies of Sr-90. Future sump dilutions and pumpouts will tend to deplete the sludge layer further, and leaching from the concrete would be evidenced if Curie balances in the sump water indicate the appearance of activity after the sludge layer is depleted.

ANALYTICAL MODEL

An attempt to model the phenomena occurring in the reactor building sump is described by equation (1) below:

$$\frac{dQc}{dt} = -\lambda Qc + LCr + DA\,(Qo/Vo - Qc/V) \quad (1)$$

Qc = Total activity in containment sump - Ci
Cr= Activity concentration in primary system Ci/gal
V = Volume of containment sump water - gal
L = Leakage rate from primary system into containment- gal/min
λ = Decay constant for activity - min^{-1}
D = Transport coefficient for activity in sludge/concrete to containment sump water $Ci\text{-}min^{-1}\ ft^{-2} - (Ci/gal)^{-1}$
A = Area of sludge/concrete surface - ft^2
Qo/Vo=Activity concentration in sludge/concrete surfaces Ci/gal

For Cs-137 the decay term can be neglected and after 7/15/82 the primary system leakage rate into the containment sump can be considered to be zero.

If we also assume that as activity leaches from the sludge/concrete surfaces it is replenished from within the sludge/concrete so that Qo/Vo is constant with time, then equation 1 becomes:

$$\frac{dQc}{dt} = DA\,\frac{Qo}{Vo} - DA\,\frac{Qc}{V} \quad (2)$$

defining $Ko = DA\,\frac{Qo}{Vo}$, $K = DA$

$$\frac{dQc}{dt} = Ko - K\,\frac{Qc}{V} \quad (3)$$

Integrating eq (3) gives:

$$Qc = \frac{Ko}{K} V \left(1-e^{-\frac{K}{V}t}\right) \qquad (4)$$

The concentration of activity in the containment sump is than

$$\frac{Qc}{V} = \frac{Ko}{K} \left(1 - e^{-\frac{K}{V}t}\right). \qquad (5)$$

If it is assumed that all of the Curie addition to the sump water is from the sludge layer only and that the measured Curies in the supernate from the 6/24/82 sludge samples represent an equilibrium condition, then the Ko/K term of equation (5) can be evaluated.

For Cs-137 $\frac{Ko}{K}$ = 150 uCi/ml and for Sr-90 $\frac{Ko}{K}$ = 6.93 uCi/ml

and equation (5) becomes:

$$\frac{Qc}{V} = 150 \left(1 - e^{-\frac{K}{V}t}\right) \text{ for Cs-137} \qquad (6)$$

$$\frac{Qc}{V} = 6.93 \left(1 - e^{-\frac{K}{V}t}\right) \text{ for Sr-90} \qquad (7)$$

Applying equation (6) for Cs-137 concentrations between pumpouts 17 and 18 when the water volume in the sump remained relatively constant over a period of 37 days and solving for $\frac{K}{V}$ results in equation (8):[6]

$$\frac{Qc}{V} = 150 \left(1 - e^{-\frac{t}{127}}\right) \text{ for Cs-137} \qquad (8)$$

where t is in days

Equation (8) indicates that equilibrium is approached slowly and full equilibrium would require almost 2 years. This rate would be too slow for practical decontamination. Additional data from pumpouts could confirm the rates and help define the practicability of leaching as a technique for decontamination of the El. 282' 6". It should be noted that as the source becomes depleted equations 6, 7, and 8 will no longer be valid as the assumption that Qo/Vo remain constant with time is no longer true.

IDENTIFICATION OF ACTIVITY SOURCES

In an attempt to identify other possible source of Cs-137 in the 282'-6" El. Table 1 was generated. It indicates other probable major sources in addition to the sludge layer. These should be the subject of specific decontamination action plans. It is noted that the reactor building liner surfaces are probably small reservoirs of contamination.

TABLE 1

ESTIMATED CURIES IN CONTAMINATED STRUCTURES
ON EL. 282-6

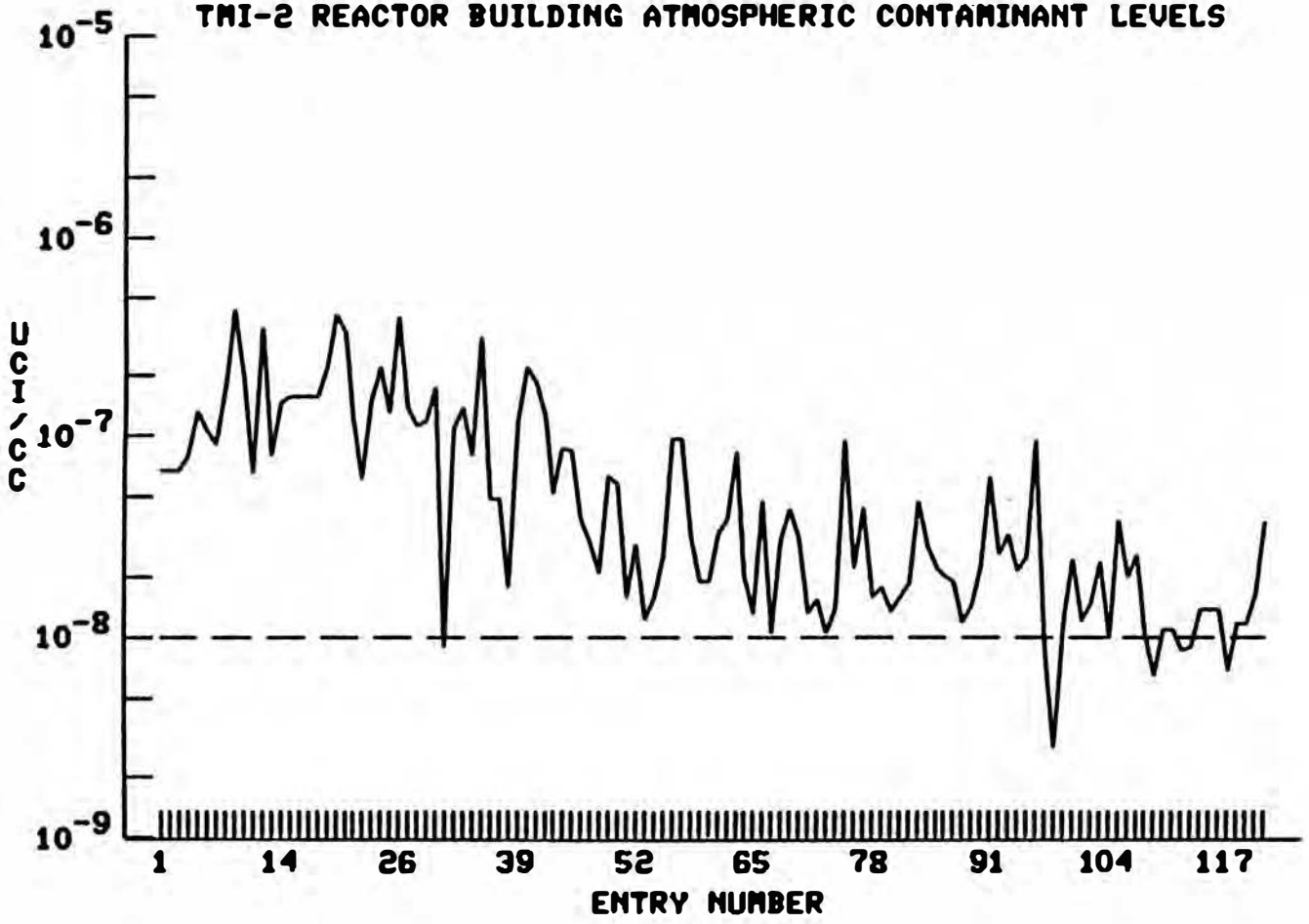
Contaminated Structure	Estimated Cs-137 Conc. uCi/ml (Ref.9)	Estimated Total Curies Cs-137 (Ref.8)
Enclosed Stair Concrete Blocks Fully Flooded to El. 291-10	102	1281
Concrete Fill Slab (Cast)		
Assume Penetration 2 in.	8	386
6 in.	8	1158
24 in.	8	4631
Vertical Concrete Walls (Cast)		
Assume Penetration 1 in.	8	196
2 in.	8	393
Containment Liner	30 uCi/cm^2 (Ref. 7)	106
Sludge		
Assume Slurry 3/4 in. deep		
Total Slurry	204 (Ref.4)	3672
Soluble Solids	187	3360
Insoluble Solids	17	312
Reactor Coolant Drain Tank	50 (Ref.7)	1370
Water at Bottom of Elevator Shaft	160	2569
Concrete Block Baffle Wall Assume Saturated	33	110

ATTACHMENT IV-4

PREPARED BY: L R Frey

November 18, 1982

TMI-2 REACTOR BUILDING ATMOSPHERIC CONTAMINANT LEVELS



SOLID LINE : AVERAGE BZA CS-137
DASHED LINE : MPC 1.0E-8 UCI/CC (INSOLUBLE) 40 HOURS PER WEEK

11/18/82

V.A PLENUM REMOVAL AND UNDERHEAD EXAM

Babcock and Wilcox, in association with GPUN/Bechtel personnel made two presentations to TAAG relative to the draft of "Three Mile Island-Unit 2 Planning Study for Plenum Assembly Removal". During these presentations, TAAG comments were made. The following discussion summarizes these comments and recommendations relative to plenum removal:

1. Plenum Assembly Axial Distortion Evaluation

   B&W evaluated the effect of radial temperature distribution on radial distortion of the plenum. Large temperature gradients are believed to have existed at the time of the accident and may have also produced axial distortion of the plenum assembly.

   Recommendation:

   It is recommended that an evaluation be made on the effects of having the control rod guide assemblies near the center of the plenum assembly at a higher temperature during the accident than the plenum cylinder and the control rod guides at the periphery of the assembly. Such a temperature distribution would produce thermal displacements in the axial direction thereby placing the bolts joining the plenum lower grid to the plenum cylinder in tension. Bolts at the lower end of peripheral guide tube assemblies would also be placed in tension. Bolt yielding might cause the lower grid to move downward, and perhaps interfere with the core support shield bottom flange.

   B&W agreed that such an evaluation should be performed, but believe that it unlikely that it will show any adverse effects. In particular, (1) there was no evidence during the "Quick Look" experiments that the plenum lower grid had moved downward, and (2) the flow path within the plenum is across control rod guide assemblies to the hot legs, so that substantial radial temperature gradients to the assemblies are unlikely.

2. Predicted Gamma Dose Rates at the Work Platform and During Dry Plenum Removal

Predicted gamma dose rates are based on the B&W assumption that an 0.1 inch thick layer of core debris is present on the plenum top cover and a 1 inch thick layer of core debris is present on surfaces at the bottom end of the plenum assembly. The amount of debris at intermediate surfaces was assumed to vary linearly from top to bottom of the assembly. In this regard B&W clarified that the amount of debris assumed is an upper limit, compared to what was actually seen during "Quick Look" experiments, and the assumed layer thicknesses were used to conservatively estimate the gamma dose rates.

Recommendation:

B&W should provide recommendations (at GPUN/Bechtel's request) for any extensions to the currently planned radiation measurements above the plenum cover within control rod guide assemblies. In particular, B&W should assess whether additional measurements within the plenum assembly would be of benefit for plenum removal.

3. Airborne Contamination Associated with Plenum Removal Operations

During the two aforementioned presentations on plenum removal, several TAAG comments were directed towards possible reconsideration of thoughts and actions regarding the control of airborne radioactivity. A summary of these comments is provided below.

a. Particulate Contamination

TAAG considers the major potential airborne hazard during plenum removal will be from Cs-137 as has been experienced previously in reactor building work and not from Sr-90 as stated by B&W. The hazard is real and should not be underestimated but it is not considered that extraordinary means, other than routine containment techniques are necessary to control airborne radioactivity.

Recommendation:

The use of special techniques is not considered necessary and should be avoided. Such special techniques include dry inert purges; maintenance of negative pressures within the plenum container, "canning" of the plenum upon removal from the reactor.

b. Kr-85 Releases

The B&W study assumes that ten percent of the Kr-85 gas remaining in the fuel might be released during plenum removal, and states that this is a matter of concern which requires more detailed evaluation. In this regard:

Recommendations:

(1) TAAG considers that the assumed Kr-85 release may be unrealistically high. B&W stated this assumed release was not based upon a mechanistic evaluation of fission product release from fuel. TAAG considers that a mechanistic evaluation should be performed to define more realistically the amount of gas release which could occur.

(2) Even in the event that the releases are as high as presently predicted, they are not considered to be a major difficulty for plenum removal. Various alternatives are available to protect personnel from the airborne radioactivity such as:

(a) use of supplied air to provide higher protection factors.

(b) limiting stay times until containment purging lowers the activity to acceptable levels.

(c) use of a partial cover over the indexing fixture to the reactor building ventilation system to assure capture of the majority of the released gasses.

4. Cesium Spikes During Plenum Lift

TAAG considers that additional consideration should be given to the possibility of large cesium spikes created during initial plenum lift.

Recommendation:

Included in plenum lift procedures should be an operational hold after jacking of the plenum approximately at the 1/2" point and perhaps at other intervals such that measurements can be taken to assure that a cesium and krypton release has not occurred under the plenum due to disruption of either intact or damaged fuel.

5. Destructive Versus Intact Plenum Removal

B&W is proceeding with plans to remove the plenum as an intact assembly in essentially the normal fashion. As a contingency, however, at the request of GPUN/Bechtel, B&W is also evaluating destructive plenum removal, i.e., cutting it up in-place and removing it in segments, and planning to perform a comparison evaluation of intact versus destructive removal.

Recommendation:

TAAG considers that the correct approach is to use the normal type of intact plenum removal, with appropriate plans to park the plenum in a safe condition in the event of unexpected difficulties. Contingency planning for destructive removal and a comparison evaluation of intact versus destructive removal are not considered warranted. Such contingency planning will add unnecessarily to the plenum removal effort, and probably be unrealistic with regard to destructive removal.

6. Dry Versus Wet (Canal Flooded) Plenum Removal

B&W has recommended that plenum removal be performed with the canal dry, instead of with the canal flooded, for improved access to perform various operations. A hybrid approach identified by B&W is to perform the initial assembly lift (about 10 inches) with the canal dry and perform the remaining lift and transport to storage with the canal flooded.

Recommendations:

TAAG considers this hybrid approach could be attractive because it could simplify plenum removal, e.g., eliminate the need to bag the assembly until removal from reactor. TAAG recommends that further consideration be given to this hybrid approach, although further extensive evaluations are not considered necessary.

7. Work Platform Location

It is currently planned that the indexing fixture top cover will serve as the work platform during plenum removal. This may require a very complex cover which, in addition to serving as the RCS seal, will contain shielding, access holes for various tasks and operations, etc.

Recommendations:

TAAG suggests that an alternate approach be considered which would involve a very simple sealing cover and a separate work platform located above the indexing fixture.

8. Initial Lift with Jacks Instead of the Polar Crane

B&W plans to lift the plenum assembly through the first 10 inches using jacks rather than the normal crane. TAAG agrees with this approach, particularly in view of several factors which were not mentioned in the draft study report, i.e., (1) if the crane were used, the rigging would interfere with access above the indexing fixture required for various inspections and operations during the initial lift, and (2) if lifting operations were delayed significantly, jacks would avoid having the plenum suspended from the crane for a long period of time.

9. Plenum Cleaning

The B&W plan involves use of a vacuum cleaner which discharges any collected debris back into the reactor vessel in the core area. This may stir up the core debris and prevent effective plenum cleaning.

Recommendation:

TAAG recommends that an alternate flushing technique be employed which would flush the plenum as it emerges from the water; allowing the flush water to flow down into the reactor vessel.

10. Index Fixture Key Removal

The keys located within the indexing fixture are a possible inference during plenum lift. Such an inference could cause the index fixture itself to lift during plenum removal, and its contained water to leak out.

Recommendation:

TAAG recommends that the keys be removed prior to the plenum lift test.

11. Auxiliary Crane

The B&W study indicates that the auxiliary crane will be employed for various operations prior to plenum lift. The auxiliary crane is not being refurbished, and will not be available.

Recommendation:

TAAG recommends that an alternative to the auxiliary crane be identified.

V.B UNDERHEAD EXAMINATIONS

TAAG has reviewed the requirements for performing under-the-head examinations of the potential effects of fuel debris accumulation on the inside surface of the reactor vessel head and the top cover of the plenum. The results and recommendations of the TAAG review on this matter are summarized as follows:

1. Recommended Under-the-Head Exam

   a. TAAG recommends an under-the-head gamma scan via a leadscrew hole be performed to establish whether the gamma radiation levels during head removal will be within the levels assumed by Bechtel in planning the head removal. Even though the information available to date indicates there should be no significant fuel debris on the plenum top cover (see Section V.B.3 below), the plans assume that some fuel debris may be present on the plenum top cover. In particular, the Bechtel plan assumed a gamma radiation level of 8.5 r/hr with the head removed at a point four feet beyond the vessel inside diameter and five feet above the vessel flange surface.

   b. The proposed gamma scan will also be useful in determining whether gamma radiation levels are significantly greater than the radiation levels experienced during head removal at normal plants. Such radiation levels at B&W plants are typically on the order of 3 r/hr below the head at the bottom of the leadscrew support tubes, and have been as high as 50 r/hr. Radiation levels on the exterior surface of the head are typically on the order of 200 to 300 mr/hr. With the head in place on its storage stand, doses beyond the periphery of the head at waist level have been on the order of 5 r/hr, i.e., comparable to the dose employed for planning purposes at TMI.

c. Examinations are not considered necessary to determine beta or alpha radiation levels. In particular, Bechtel's plan calls for fully shrouding the head and to cover the reactor vessel opening when the head removal operations are complete. Under these conditions, alpha radiation would be absorbed in the plastic bag type of shroud, and would not represent a source of exposure to personnel. The head shroud called for by the Bechtel head removal plan should also control airborne contamination during head removal operations. In this regard, it is noted that breathing zone apparatus (BZA) measurements during initial gas venting for the "Quick Look" examination, when significant quantities of gas e.g., (500 to 1000 ft^3 STP) were vented from the reactor vessel to the containment building, indicate that airborne contamination, including alpha, was not excessive.

2. Details Regarding the Proposed Gamma Scan

a. It is recommended that the proposed gamma scan be made underneath the reactor vessel head by inserting a radiation detector down through the CRDM's which had the leadscrews removed during the "Quick Look" examinations. Such a measurement could be performed with the vessel water level at its current elevation (e.g., about 333 ft. elevation) and with the current coolant concentration of Cs-137, i.e., about 3 μCi/ml. Based on the preliminary dose estimates summarized in Table V.B-1 such measurements should provide clear evidence of whether the gamma source on the plenum top cover is equal to or less than that assumed by Bechtel for head removal planning purposes. Preliminary calculation indicates that the dose rate limit, at an elevation near the top cover of the plenum and underneath the reactor vessel head, corresponds to about 90 r/hr (with the leadscrews in their lowered positions).

b. The radiation measurements could be obtained using a tree of TLDs. As an alternate possibility a gamma detector such as the Eberline RO7 can possibly be employed. Radiation measurements should be obtained at, as a minimum, the three different elevations defined in Table V.B-1 and in two CRDM positions. See Figures V.B-1 and V.B-2 for areas to be scanned. If TLDs are employed, they should be suitably shielded to exclude beta and alpha dose rate contribution.

3. Evaluation of Condition Under the Head Based On the Quick Look Examination & Reactor Internal Flow Characterization

a. In particular, the "Quick Look" video tapes show that the top surface of control rod guide assembly first and second support plates have only the light corrosion film typical of that found in normal plants. (See Figure V.B.3) This indicates that the plenum cover should also be free of debris. The "Quick Look" tapes also show the bottom surface of each support plate is not contaminated, which indicates that there is no reason to believe there is any debris on the inside surface of the reactor vessel head.

b. These results of the "Quick Look" examinations also appear reasonable, based on the flow conditions predicted to exist at the time of the accident. In particular, the principal means by which fuel debris could reach the plenum top cover and inside surface of the vessel head is by entrainment in fluid flowing upward inside of the control rod guide assemblies. With one reactor coolant pump running such as occurred after the March 29, 1979 accident, the vertical velocity within a guide assembly is estimated to be in the order of 0.3 feet per second in the region between support plates, and about 0.45 feet per second at the elevation of the support plates. This velocity is low enough to permit entrained fuel debris to settle out before it could reach the top end of a guide assembly.

c. Based on the above evidence and data there should be no debris other than the normal light type corrosion film on the inside surface of the reactor vessel head or on the top surface of the plenum. The gamma scan recommended by TAAG in Secton V.B.2 will provide further confirmatory information in this regard.

4. Conclusions Regarding Under-the-Head Examination

a. The gamma scan recommended in V.B.2 should establish whether the gamma radiation levels during head removal will be within the levels assumed by Bechtel for head removal planning purposes.

b. The gamma scan does not require lowering the reactor water level, or any processing to decrease existing Cs-137 concentration in the coolant. Accordingly, it should be possible to perform such examinations in the near future.

c. If the measured gamma radiation levels in the test proposed in V.B.2 substantially exceed those assumed for the planned head removal operation, means for cleaning the top plenum cover and the reactor vessel head may be developed.

If there are no surprises out of the above tests, TAAG sees no requirements for "under the head" examination involving removal of a CRDM or cutting of leadscrew support tube.

TABLE V.B.1

"PRELIMINARY"

ESTIMATED GAMMA DOSE RATE
INSIDE A CRDM GUIDE TUBE

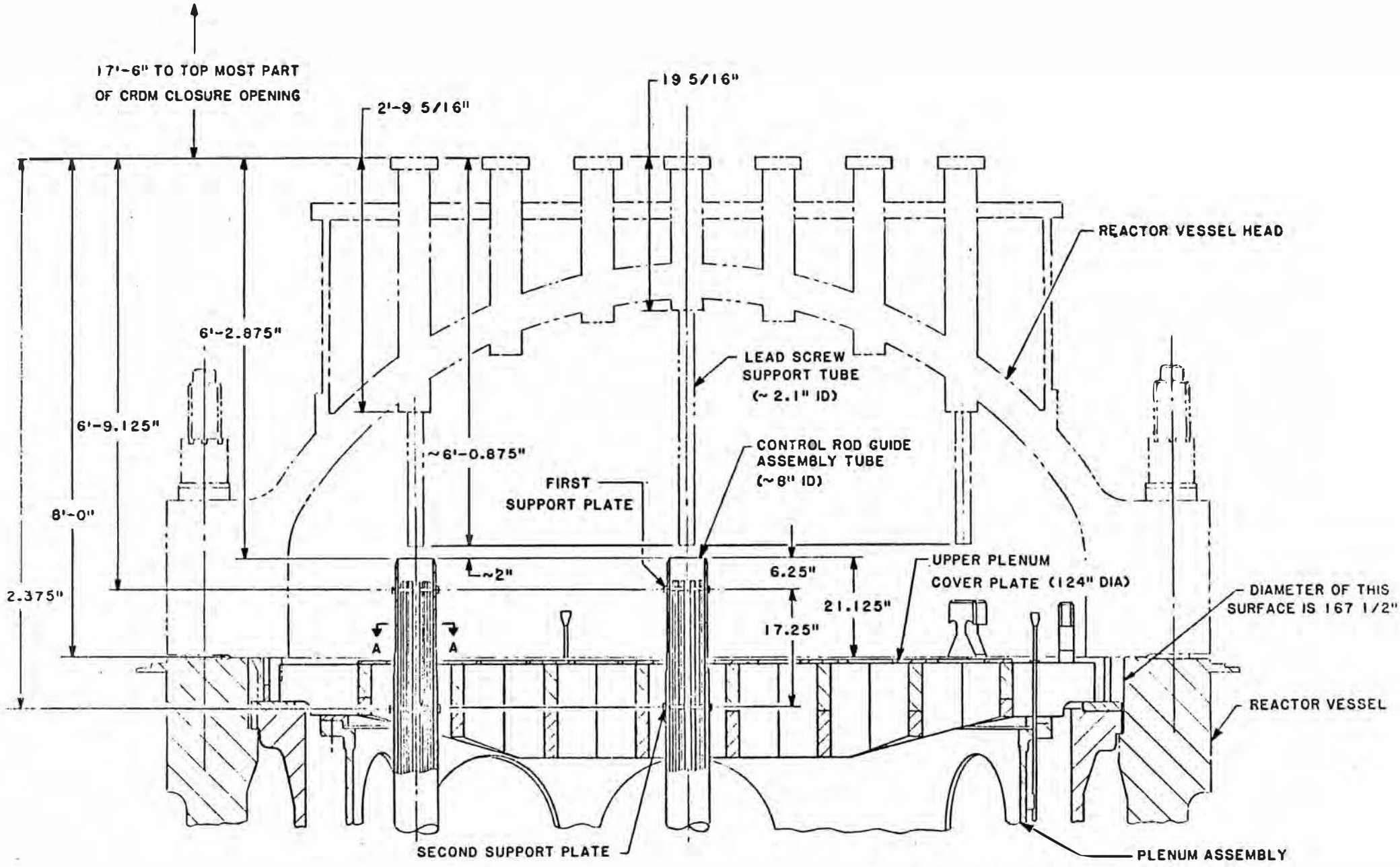
Elevation of Radiation Detector*	Estimated Dose Rate, r/hr	
	If No Fuel Debris Is Present (1)	With the Planned Amount of Fuel Debris Present On The Plenum Top Cover (2)
At the inside surface of vessel head	4	4
Mid-height of head	7	9
Top of the plenum cover plate	7	90

(1) The estimated dose is based on the head being filled with water containing 3 μCi/ml of Cs-137, and the leadscrews at their fully inserted position.

(2) The estimated dose includes a contribution from Cs-137 in water, plus a contribution from surface contamination on the plenum cover, equal to the amount assumed for planning purposes. This planned contamination is equal to the amount which would give a dose rate of 8.5 r/hr with the head removed at a point 5 feet above and 4 feet beyond the vessel inside diameter.

*See Figure V-B-1 for the areas to be scanned.

Figure V.B-1



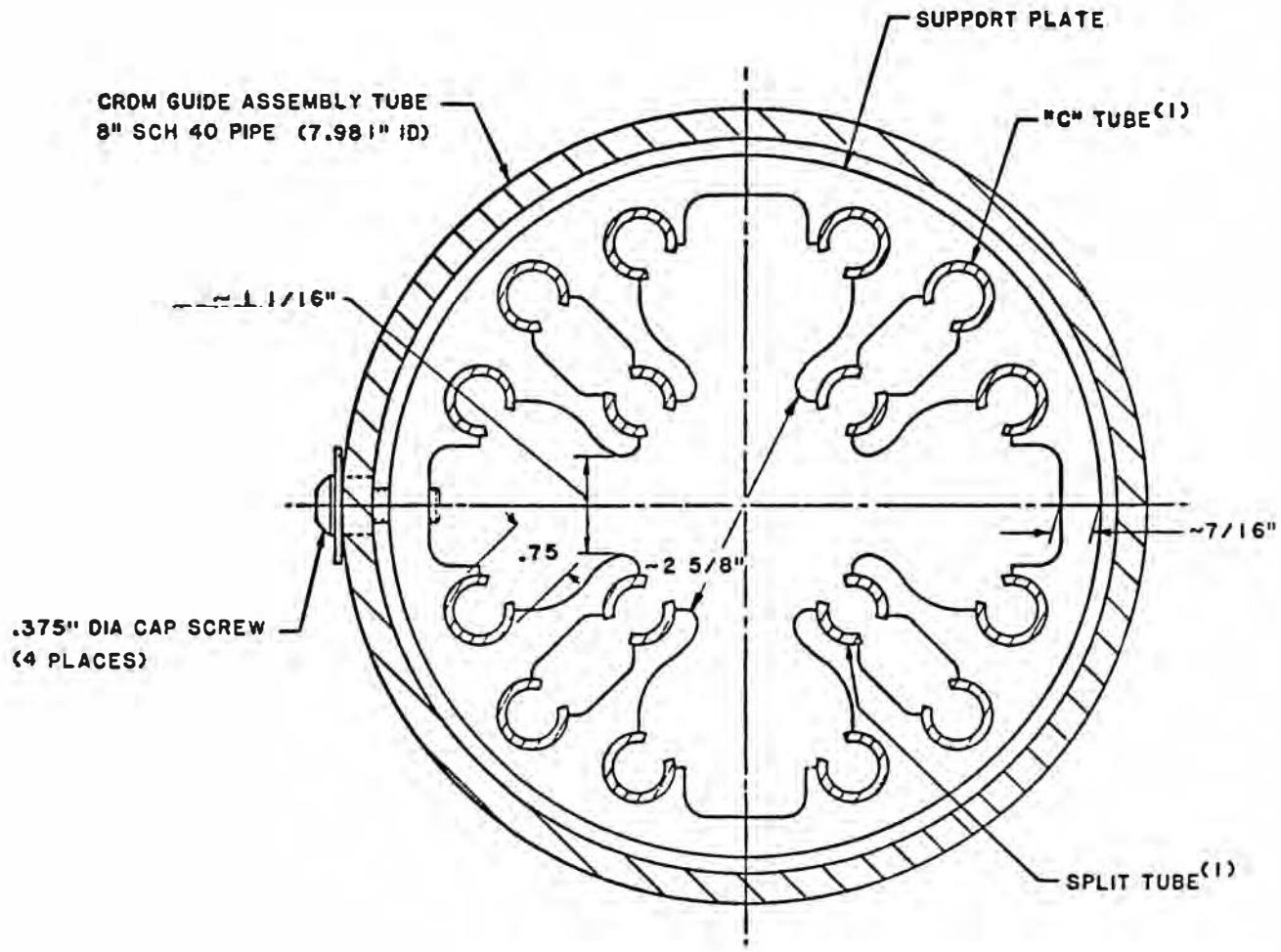
KEY ELEVATIONS FOR MEASURING UNDER-THE-HEAD RADIATION LEVELS
TO ASSESS THE NEED FOR AN UNDER-THE-HEAD
PHYSICAL INSPECTION

FOR SECTION A-A
SEE MPR SKETCH SK-1074-01-519

MPR SKETCH
SK-1074-01-518
REVISION 0

Figure V.8-1

Figure V.B-2



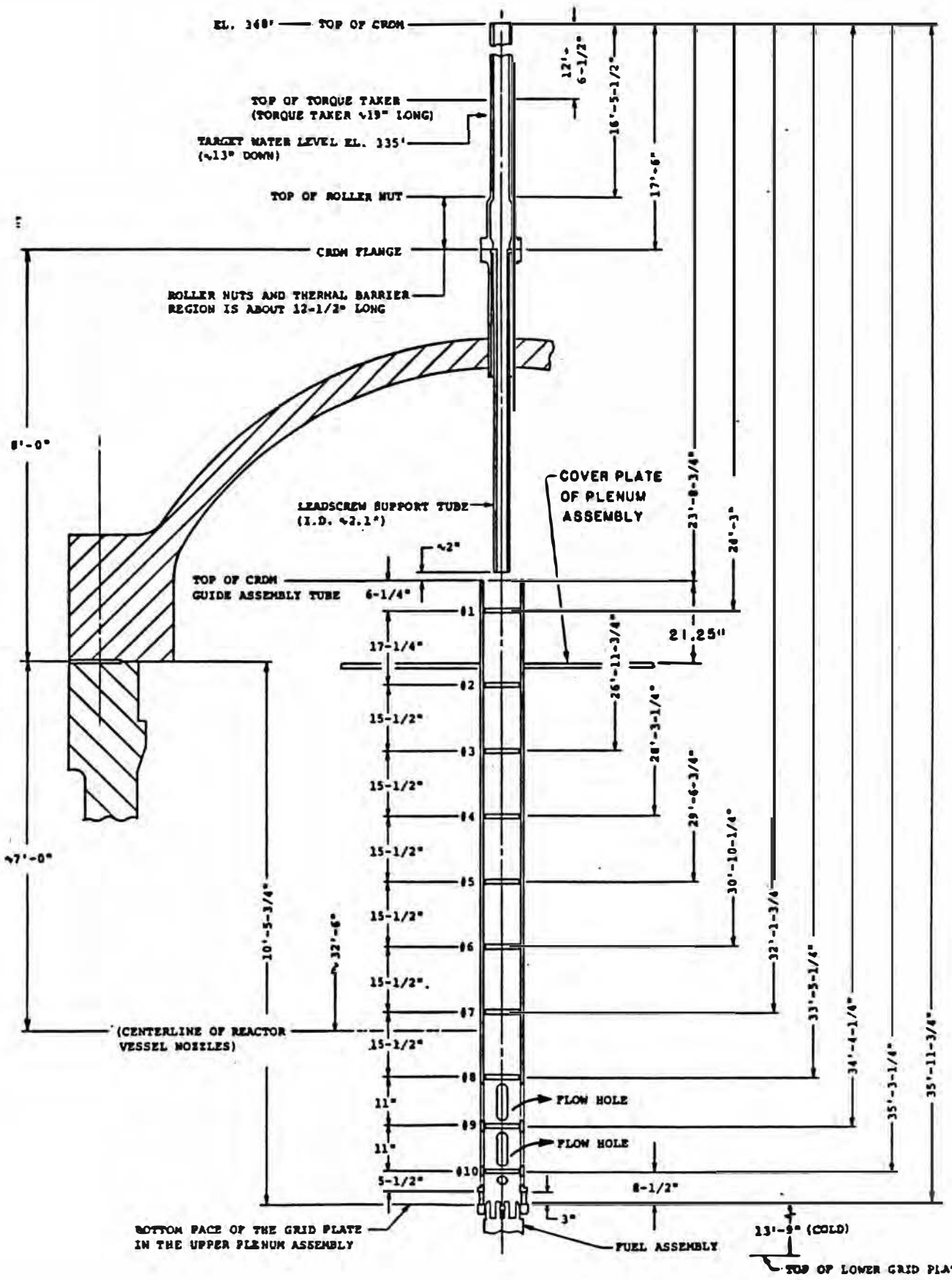
(1) BOTH OF THE "C" AND "SPLIT" TUBES ARE BRAZED INTO EACH SUPPORT PLATE

SECTION A-A
TYPICAL SUPPORT PLATE
OF CONTROL ROD GUIDE ASSEMBLY

MPR SKETCH
SK-1074-01-519
REVISION 0

Figure V.8-2

Figure V.B-3



MPR ASSOCIATES
SK-1074-01-512
REV. 1
5/12/82

DEPTH OF VARIOUS PARTS TO BE INSPECTED VIA A TV CAMERA INSERTED DOWN A LEADSCREW HOLE IN TMI #2

Figure V.B-3

VI. PREREQUISITES FOR HEAD REMOVAL

The "Third Report of the Technical Assistance and Advisory Group (TAAG) dated August 31, 1982 reported the results of TAAG's examination of the prerequisites for early lifting of the reactor vessel head. During the current reporting period TAAG has pursued with GPU/Bechtel their implementation of the TAAG prerequisite recommendations.

In most cases GPU/Bechtel concurred with the listed prerequisites. In a few cases GPU/Bechtel indicated that the prerequisites are not applicable and TAAG concurred. In two cases GPU/Bechtel believed additional study was required to confirm the appliability of the TAAG recommendations. Agreement was not achieved in two cases concerning proposed prerequisites.

A. Based upon the above TAAG comment, GPU/Bechtel discussions it is considered that the following prerequisites for headlift listed in the "Third Report of Technical Assistance and Advisory Group (TAAG)" are no longer considered applicable.

Prerequisite C7:

The possibility of disturbing hydrogen pockets in the head volume during head lift should be considered. It may be desirable to maintain a nitrogen purge between the time the water level is lowered until the head lift is started. It may also be desirable to reestablish a nitrogen cover over the core after the cover plate has been installed.

Discussion: GPU/Bechtel indicated that because the plant has been vented it would not be necessary to take special precautions. TAAG agreed.

Prerequisite 1N:

If it can be accomplished without interfering with the schedule for the preparation of the report, the Safety Report should cover the safety of the removal of the plenum. This evaluation would be limited to an essentially normal removal process and would not include an evaluation of a contingency removal process, such as cutting up the plenum in place.

Discussion: GPU/Bechtel indicated that the preparation of the safety report for the head lift had progressed to the point where it was not possible to incorporate plenum removal considerations without delaying the issuance of the report.

Prerequisite B11:

A cover plate should be provided to cover the pressure vessel opening. It is considered desirable, but not mandatory, that this cover plate should be leak tight or be able to control leakage in the event the primary system is refilled (unpressurized). This cover plate should have provisions to sample continuously the primary coolant and have inspection ports. This plate should be designed to be brought in through the personnel access hatch.

Discussion: GPU/Bechtel indicated that it would not be feasible to provide a leak tight cover. In addition, the indexing fixture planned for use after head removal will not be leaktight for heads of water greater than 6 or 8 feet. GPU/Bechtel considered that such a cover would not be necessary to provide suitable casualty control. TAAG concurred.

Prerequisite D1:

Radiation levels in the areas where work is to be performed should be controlled to less than 50mr/hr. These levels should be achieved by controlling water activity, through the use of shielding, or some combination of these two factors.

Discussion: GPU/Bechtel indicated that based upon work to date, it may not be feasible to achieve radiation levels below 200 mr/hr in time for this scheduled head removal operation, depending on the results achieved by the dose reduction task force.

B. Additional studies are planned which will relate to the feasibility of the following prerequisites:

Prerequisite: B12:

"Prior to the removal of the head, the seal plate should be installed. It is recommended that the plate design be revised to provide a higher probability of a leak tight seal that could be relied upon for a period of several years. An all welded installation is preferred."

Discussion: GPU/Bechtel agreed that the plate should be installed prior to head lift but did not agree that an all welded installation is required. Studies are planned to evaluate an improved seal design. The results of these studies will be used to determine the need for a welded seal.

Prerequisite B1f:

"The potential loss of coolant accident should be re-evaluated for head lift and post head lift conditions. If possible, it should be shown that a "dry" core does not present problem."

Discussion: GPU/Bechtel indicated that there may not be any advantage to showing that a dry core would not present a problem. They agreed to evaluate the possible advantages.

C. Agreement was not reached with respect to the following prerequisite.

Prerequisite C3:

"An enclosed environment with a clean air source should be provided to minimize the need for respirators and simplify contamination control."

GPU/Bechtel Comment: No plans are being made to provide an enclosed environment. Respirators will be used.

TAAG Response: TAAG considers that this recommendation has not received an adequate evaluation and represents an important factor in the ALARA aspects of the head lift and subsequent operations.

Prerequisite 2d:

Consideration should be given to include the use of commercially available direct alarming boron monitoring equipment to monitor the boron concentration in the pressure vessel after head removal. This may simplify other water inventory monitoring requirements.

Discussion: GPU/Bechtel indicated that the commercially available direct alarming boron monitoring equipment was not qualified for use with contaminated primary coolent. As a result they considered the use of this equipment unproven and that it could potentially cause problems.

TAAG Response: It is understood the CE is manufacturing equipment qualified for use with radioactive coolant. The use of the CE equipment should be evaluated.

VII. SDS AND CANAL WATER SYSTEMS

RECOMMENDATIONS

1. The recommendations in the previous TAAG report (third) remain valid.

2. All discharge water from the underwater vacuum should be processed for fission product removal to minimize the effects of any increase in leach rate during defueling.

INTRODUCTION

The Third Report of the Technical Assistance and Advisory Group (TAAG), dated August 31, 1982, included a comprehensive discussion of the control of water quality (Chapter III) and seven specific recommendations. Based on information supplied to TAAG some aspects of water processing have been considered, but relatively little has been accomplished during this report period. Specific decisions for future action have not been identified. Other, more visible tasks appear to have diverted the overall guidance which is required to bring about an integration of the several systems (such as SDS, vacuum system, canal water filtration, canister design, waste handling, and system interconnections) and to proceed. TAAG believes Chapter III of the previous report still provides a good basis for proceeding and recommends that greater emphasis be placed in this area.

STATUS OF PRIOR RECOMMENDATIONS

The status of the actions taken in response to the specific recommendations in the last report, as understood by TAAG, is summarized here.

1. Examine existing equipment in the spent fuel cooling system for applicability. A preliminary review of this equipment has been made, and it was concluded that some of it is useful. Decisions on which components should be replaced or the nature of the modified system remain to be done.

2. SDS upgrade. The column in the SDS system, itself, can be interconnected in any pattern and operation with four parallel lines of two columns each, each line at up to 15 gpm flow, appears practical. There has been evaluation of the installation of a pump in the canal or reactor vessel (following head removal) and a line to carry water to the SDS, but the system under consideration would have a flow capability of probably not more than 30 gpm and would supply two SDS lines. There appears to be concern that a higher capacity cannot be justified without further information on the source term.

   It is difficult to conceive of a situation in which excess water processing capability will be available, since higher processing rate translates directly into lower radiation exposures. Furthermore, the average through put will be well below the maximum, and the average should exceed 30 gpm. The capacity recommended in the previous report is believed to be approximately the maximum for effective operation with the existing SDS components. TAAG therefore recommends that a capacity of 60 gpm with four parallel SDS lines should be the immediate design basis.

3. High-capacity backup system for SDS. Some preliminary consideration may have been given to using Epicor-2 liners loaded with zeolites for this function, but at a lower throughput than recommended.

4. Interface of vacuum system into water processing system. There was no indication that this is being specifically addressed. TAAG believes this is an important area, and further consideration is provided below.

5. Interconnection for water handling systems. Some examination of existing systems has been made, and a new supply for SDS from the reactor vessel region is under consideration (see 2). An integrated approach to the design of the water handling systems apparently has not been initiated.

6. Possible advantage to use of a barrier to isolate most of the canal from the reactor vessel. Nothing was mentioned in this regard.

7. Develop processes to remove deleterious chemical impurities from canal water. There has been no action on this.

FURTHER CONSIDERATION OF VACUUM SYSTEM IN REGARD TO SOURCE TERM.

There has been some delay in establishing design goals for water cleanup partly because of uncertainty about the source terms (leach rate for soluble radionuclides and suspension rate for insolubles). The source term will not really be known until the actual fuel removal operations are undertaken, but the cleanup system has to be in place before that. Fortunately, the source term for soluble activity need not be an important variable if the proper water management scheme is employed; and to some extent, the same may be true for particulates.

The source term for cesium activity over the last year was estimated to have been about 2 Ci/day, with a fairly large uncertainty. A more precise estimate probably can be obtained when the RCS is refilled and when further SDS processing is done. It was shown in the previous report that a source term of this magnitude can be dealt with (the canal water can be maintained below 0.1 μCi/ml by processing at a realistic rate). The concern, then, is that manipulations required for fuel removal will cause a greatly increased source term from debris disturbed by defueling operations.

This can be prevented, however, if the water in contact with that debris is not allowed to mix into the canal water, but is preferentially withdrawn into the zeolite ion exchange cleanup system (SDS). The effluent cesium concentration from SDS will be well below that of the canal water, regardless of the influent concentration; thus, any such increase in source term would be immediately removed so it would never reach the canal water. Since degraded debris will also be picked up along with the water, and collected, any increased leaching from that will also be processed by the cleanup system. To achieve this result, two steps are necessary.

First, the underwater vacuum system should be running whenever any core removal operations are underway, and it should take suction from the immediate vicinity of those operations. Then, any broken fuel or fines would be drawn into the vacuum, along with water exposed to any new surfaces (which are the anticipated source of increased leach rate). If there is increased fission product leaching, then, the activity would be largely contained within the stream flowing to the vacuum or originate within the vacuum, rather than being dispersed into the canal water.

Second, the entire flow discharged from the vacuum should go through zeolite ion exchange processing (such as SDS), where it will be decontaminated and returned to the canal. The majority of any material or surface with increased leach rate should be carried into the vacuum and removed with the collected debris, so any increase in leach rate from the remaining core, when removal operations are interrupted, should be relatively small. As a result, even a substantial increase in leach rate, as long as it is localized in the region of core removal operations or largely associated with material carried into the vacuum system, should have very little effect on the canal water activity.

There may be a mismatch between the flow rates of the vacuum and the ion exchange system, but this can be dealt with by providing surge capacity between them. The SDS flow should average somewhere between 30 and 60 gpm, and a larger capacity system could be devised if necessary. (This relates to waste generation, which is directly proportional to the volume of water procesed, and not to the Curie content.) The flow requirement for the vacuum system has not been established, but it might be within this range, or it might be larger. Since defueling operations will be intermittent and SDS operation continuous, the vacuum flow rate could be several times larger than the SDS rate if adequate surge capacity is provided.

It is apparent that, with this mode of operation, the vacuum discharge must be well-clarified prior to going to SDS. The design of the debris collection and filtration system for the vacuum requires very careful evaluation. Since it may contain surge tanks for other reasons, it may be advantageous to use them as sedimentation vessels for removal of all but quite small particles, thereby reducing the solids load on the filters for clarification of this vacuum effluent - SDS feed. (There will be a seperate and larger filter system for canal water containing very low solids concentrations.)

Considerations such as the foregoing reinforce the recommendation that the vacuum system should be designed as an integral part of the water decontamination system, that the design capacity of SDS should be based on its maximum practical flow rate, and that all associated systems should be sized accordingly. TAAG therefore recommends that a design flow capacity of 60 gpm should be installed at the outset.

VIII. MAN-REM EXPOSURE ESTIMATES

A. Introduction

The "Third Report of the Technical Assistance and Advisory Group (TAAG)", dated August 31, 1982, discussed the ALARA exposure estimates associated with the cleanup. It was noted that difficulty was observed in decontaminating to achieve target dose rates and that this fact could have a substantial effect on actual occupational exposures. This increase in exposures could result in total man-rem levels in excess of the NRC "Programmatic Environmental Impact Statement" (PEIS). It was concluded that early identification of any differences from the PEIS estimate would be desirable and that TAAG should evaluate the reactor building occupational exposures received to date, to determine if the PEIS estimate is applicable.

During the period of this report, TAAG performed the above evaluation. Discussions were held with NRC and GPU/Bechtel relative to the basis used to establish the PEIS estimate and the experience to date with the reactor building activities. This section of the report summarizes the results of this review.

TAAG was assisted by Mr. Glenn Hoenes of Pacific Northwest Laboratories in their efforts to review the man-rem estimates for the TMI-2 cleanup. The following discussion is extracted, in part, from the draft report he prepared summarizing the results of his and TAAG's reviews.

B. Conclusions

As a result of their review, TAAG concludes that the PEIS man-rem estimates of from 2000 to 8000 man-rem for the defueling and cleanup are low and that the actual man-rem level will exceed 10,000 man-rem. This increase in predicted man-rem exposure is a result of the following considerations:

1. The strategy factor used to account for the effects of working in lower than average radiation fields, use of shielding, positioning workers in low radiation fields and other radiological engineering considerations was overly optimistic.

2. Estimated expected average radiation levels of 10 milli-rem are more likely to be in the range of 50 - 60 milli-rem.

3. The sequence of work was different than initially planned with a larger amount of work to be accomplished prior to decontamination activities.
4. The decontamination activities have not resulted in the anticipated reduction in dose levels.

Although the anticipated exposure levels may be larger than predicted, they do not represent a potential exposure to the population as a whole, but are limited to the work force. The average operating reactor plant total worker radiation exposure is about 800 man-rem per year. The average exposure to date for TMI-2 cleanup has been about 200 man-rem per year. The results of this TAAG review indicate that the average exposure levels for the remaining period of the cleanup will increase and be larger than that occuring at operating plants. However, this difference will be significantly less than an order of magnitude and will be less than or about the same as the 2000-3000 man-rem per year levels associated with major maintenance operations at these operational plants.

C. DISCUSSION

This presentation will summarize a comparative analysis and appraisal of the occupation dose estimates for TMI-2 recovery made by Bechtel for General Public Utilities and those made in the PEIS by Argonne National Lab for the Nuclear Regulatory Commission. The total occupational doses estimated by the two agencies vary considerably (see Table 1). The estimates made by ANL are a factor of 3-4.5 lower than those made by Bechtel.

Most of this difference is a result of estimates made for one step in the recovery operations: decontamination of the reactor building. Estimates for this task made by ANL are a factor of 7 to 10 lower than those made by Bechtel. This discrepancy could be attributed to large differences in assumptions about time needed to accomplish the job or about radiation exposure rates. This possibility was investigated. The purpose of the analysis was to determine and evaluate the basic differences in the estimates.

In a presentation made to T.A.A.G. on 22 September 1982, the NRC spokesman referred to an estimate of 171,000 man-hours to accomplish the reactor building decon. On that same day, a representative of Bechtel showed estimates of 142,550 man-hours for the same task. These estimates are not substantially different and are obviously not the cause of the large discrepancy in dose estimates. The PEIS contains estimates of man-hours ranging from 300,000 to 900,000 to accomplish the decon task.

Radiation exposure rates used to make the estimates are listed in Table 2. Although differences can be seen between the assumptions made by ANL and Bechtel, they are not large enough to account for the discrepancy in dose estimates. Because of differing assumptions regarding work sequence and procedures, the exposure rates assumed by the two agencies are not directly comparable; the comparison in Table 2 shows only that there are no large differences.

Tables 3 and 4 lists the steps by which Bechtel and ANL, respectively, estimated the total occupational dose for the reactor building decon. As can be noted from these tables, no significant difference exists for

TABLE 1. Comparison of PEIS With Bechtel Estimates
(man-rem)

	PEIS	Bechtel
Expended through 8/81	--	1,300
Maintenance of Reactor in Safe Condition	8	250-350
Decon of Auxiliary and Fuel Handling Buildings	375-550	100-150
Reactor Coolant Inspection Removal of RPV Head and Internals Core Examination and Defueling	780-2,400	350-1,100
Decontamination of Primary System Components	108-1,7400	30-90
Waste Management	113-348	390-480
Decontamination of Reactor Building	660-3,000	7,000-21,000
TOTAL	2,000-8,000	9,000-24,000
w/o Reactor Building Decon	1,400-5,000	1,100-2,200

TABLE 2. Comparison of Exposure Rates (mR/h)

NRC Presentation 9/22			Bechtel Presentation 9/22	
Semi-Remote	110-175	347'	Prep & Gross Decon	110-230
Manual	100-150		Prep & Manual Decon	30-50
Support	30-75		Support	80
Semi-Remote	140-300	305'	Prep & Gross Decon	170-230
Manual	30-275		Prep & Manual Decon	30-50
Support	30-150		Support	200
Semi-Remote	300	282'	Initial	1,000
Manual	130		After Manual Decon	10
Support	30-50			

TABLE 3. Estimates of Reactor Building Decon by Bechtel

	man-hrs	mR/h	man-rem
305' el			
Prep for Gross Decon	4,300	230	1,000
Gross Decon of Floor	600	170	100
Decon Support	1,400	80	100
Prep for Manual Decon	3,400	50	200
Manual Decon	1,100	30	30
Post Decon	8,300	10	80
347' el			
Prep for Gross Decon	2,600	230	600
Gross Decon of Floor	650	110	70
Decon Support	2,000	200	400
Prep for Manual Decon	4,500	50	225
Manual Decon	4,400	30	130
Post Decon	9,300	10	90
Contingency			
282' el & D-Rings			
	1,000	1,000	1,000
	2,000	500	1,000
	20,000	500	1,000
	37,000	50	1,900
	40,000	10	400
TOTAL	142,550		9,925
		Range	5,000-15,000
Other Assoc. Activities	50,000+		2,450- 5,650
		Total Range	7,000-21,000

TABLE 4. PEIS Estimates for Reactor Building Decon

	man-hrs	mR/h	man-rem
347' el			
Semi-Remote Decon	4,480	110-175	510- 790
Annual Decon	21,000	100-150	2,100- 3,150
Support	26,250	30- 75	800- 2,000
305' el			
Semi-Remote Decon	2,500	140-300	350- 775
Manual Decon	5,900	30-275	175- 1,650
Support	13,000	30-150	400- 1,900
282' el			
Semi-Remote Decon	4,000	300	1,200
Manual Decon	40,000	130	5,200
Support	54,000	30-50	1,600- 2,600
TOTAL	171,000		12,100-19,000

the estimates. The estimate made in Table 4 is the initial estimate made for NRC. However, this estimate was revised several times. The remainder of this discussion will focus on these revisions.

In the PEIS, lower and upper bounds for the occupational dose were estimated. The lower bound for the reactor building decon was derived from experience in the Auxiliary and Fuel Handling Buildings (AFHB). Approximately 250 man-rem were used to clean 340,000 ft^2 through September of 1980. This resulted in an expenditure of about 0.8 man-millirem per square foot. Since some portions of the AFHB remained to be cleaned, it was assumed that approximately 1.1 man-millirem per square foot would be used in the AFHB. To account for support workers needed and extrapolation to the reactor building, it was assumed that 2.2 man-millirem per square foot would be used in this building. Since the area of the reactor building is 300,000 square ft., approximately 660 man-rem would be needed to clean the building. This figure provided the lower bound for the PEIS.

The upper bound was calculated by estimating the time needed to accomplish the tasks and the exposure rates in areas where the workers would be located. The initial estimate was nearly 20,000 man-rem (see Table 4). ANL felt that some assumptions which were used in this estimate were too conservative. By revision of the occupational dose estimates for the 305' and 347' elevations, the total dose was reduced to 15,000 man-rem. However, to the ANL/NRC personnel there still seemed to be too large a discrepancy between the upper and lower bounds.

Through telephone conservations with personnel at TMI, a means of reducing the upper estimate was developed. Experience in the AFHB had shown that the actual occupational dose received was substantially lower than what would be estimated by multiplying the exposure rate by the time needed to accomplish the task. It was found to be lower by a factor of 1/8 to 1/100. For the reactor building decon, ANL chose a factor of 1/5

to be conservative. This "strategy factor" accounted for the use of shielding, positioning of workers, minimizing time spent at hot spots and other dose reducing techniques. By applying this "strategy factor," the upper bound was reduced from 15,000 to 3,000 man-rem.

For several reasons, use of this "strategy factor" does not appear to be justified. The lower bound estimate is based on AFHB experience. There is no reason to base the upper bound on the same experience. The focus of the PEIS should be an accurate portrayal of the expected or probable impacts, not a demonstration of good agreement between upper and lower bound estimates.

Another comparison will illustrate one problem with attempts to apply AFHB experience to the reactor building. During the cleanup of the AFHB, approximately 280,000 man-hours and 142 man-rem wee accrued by decontamination crews through September of 1980. The total work force expended 500,000 man-hours and acquired 250 man-rem. Based on either of these sets of numbers, the average dose rate to workers was about 0.5 mrem/h. Exposure rates measured soon after the accident in the fuel handling building ranged from 150-500 mR/h, and in the auxiliary building from 50-5,000 mR/h. This information shows that the dose rates experienced by workers are substantially lower than measured exposure rates. Worker efficiency was estimated at 30-50%. Yet a factor of 2 or 3 does not account for the difference between measured exposure rates and average dose rates received by workers.

As of September 8, 1982, ninety-one entries had been made into the reactor building. As a result of these entries, the collective occupational dose was 266.68 man-rem, based on 1234.1 man-hrs spent in containment. If a worker efficiency of 30% is assumed, the average dose rate to workers was 65 mrem/h. Exposure rates measured on the 305' and 347' elevations average 350 and 150 mR/h (based on information provided by GPU). Experience in the reactor building produced better agreement between measured exposure rates and average worker dose than in the AFHB.

Table 5 lists the average dose rates to workers which result from estimates made by Bechtel and NRC, and for experience during work conducted in the reactor building and the AFHB:

The following conclusions result from this appraisal of the occupational dose estimates.

- o NRC and Bechtel estimates are not substantially different except for the "strategy factor." Estimated man-hours and exposure rates do not differ greatly between the NRC and Bechtel.
- o All differences stem from trying to extrapolate from AFHB experience to planned operations in the reactor building.
- o It is difficult to apply AFHB experience to the reactor building because:
  - Exposure rates at the AFHB were taken soon after the accident and may not reflect the actual fields to which worker were exposed.
  - Cleanup of the AFHB occurred soon after the accident, whereas the reactor building has been contaminated for over three years.
  - Conditions in the AFHB were very different than those in the reactor building following the accident.

TABLE 5. Average Exposure for Estimates

Bechtel	35-105 mrem/h
PEIS*	70-120 mrem/h
PEIS* w/"Strategy Factor"	18 mrem/h
PEIS* Lower Bound (660 man-rem)	4 mrem/h
Reactor Building Experience	216 mrem/h
w/30% Worker Efficiency	65 mrem/h
AFHB Experience	0.5 mrem/h

* Based on 171,000 man-hrs.

The above discussion should not be construed as saying that the PEIS evaluation is "wrong" and that the Bechtel estimate or range is "correct". Indeed, this discussion is intended to emphasize the difficulty in developing an accurate estimate of exposure for cleanup. This difficulty can be illustrated by reviewing the man-rem exposure estimates for the "Quick Look" inspection in TMI-2. Table 6 presents the exposures estimated prior to the inspection. These exposure estimates ranged from an early estimate of over 1500 man-rem to 45 man-rem. The actual exposure was about 22 man-rem.

This again illustrates the difficulties that are associated with developing exposure estimates for the cleanup. The above example illustrates the advantages of careful planning and attention to radiological engineering principals. It is believed that continued emphasis on reducing man-rem will further reduce actual exposure levels. However, TAAG does not believe that levels can be reduced to the point where the original PEIS estimate can be achieved.

TABLE 6.

MAN-REM EXPOSURE FOR

"QUICK LOOK" INSPECTION IN TMI-2

I. Estimates Made of Man-Rem Exposure To Do Total "Quick Look" Effort

	Man-Rem Estimate	Date	Organization Making Estimate	Remarks
1.	404.35 to 1,617.4	Week of Feb. 8, 1982	Bechtel	Bechtel's presentation to TAAG
2.	45.1	Week of Feb. 22, 1982	TAAG (EB, Newport News, and MPR)	Page 93 of TAAG's March 1, 1982 report
3.	50 to 150	June 1982	Bechtel	Quick Look Safety Evaluation Report
4.	60	July 1982	NRC	NRC letter to GPUN of July 13, 1982

II. Actual Man-Rem Exposure To Do Total "Quick Look" Effort

21.52 Man-Rem

See Pages 3-4 in Bechtel report forwarded by Bechtel letter BLMP-0479, dated August 24, 1982.

*All man-rem exposures based on performing first Quick Look into TMI-2 (i.e., inspection of one core position).

IX. FUEL CANISTER DESIGN

One of the important pieces of equipment that will be required to defuel the TMI-2 reactor is the canister that will be used to contain the fuel materials recovered form the plant. TAAG, in its meeting with GPUN/Bechtel, has discussed the interrelationship between these canisters and the other systems/components involved in the defueling activities. As part of these discussions, TAAG was asked to review the current GPU/Bechtel design activities relating to the fuel canister.

Information concerning canister design activities that has been supplied to TAAG includes oral presentations by R. Ryder (GPUN/Bechtel) and D. Wilkins (EG&G), and Specification 13587-2-R-200 Appendix E Section 3.0, FUEL CANISTERS. The GPUN/Bechetel presentation outlined the factors that would influence a canister design. The EG&G presentation was prepared for GPUN/Bechtel to provide design guidance to a design agency.

There are a number of practical constraints on canister design, and each must be given its proper consideration. For example, some interfaces will require system modification (e.g., storage racks) and others are not yet built, and in some cases, not yet defined (e.g., vacuum system, ultimate disposition), while others are quite firm (i.e., licensiong, existing shipping casks).

TAAG notes that the canister design's starting point is that the canister, in conjunction with its shipping cask and when loaded with any anticipated material, must be licensable for shipment from TMI to the storage site (Idaho). Within this limitation, and the assumption that existing shipping containers will be used, the most inflexible constraints will need to be determined and used to establish the bounds on the design, such as shape, size, and strength. Finally the requirements and means for dealing with interfaces (such as transferring fuel debris from wherever it is into the canister, temporary storage, and disposition or long term storage) will have to be devised and merged into the overall system designs. It is not apparent that bid invitations based on the refererced specifications will lead in an efficient manner to a oesign properly considering these factors.

The preliminary considerations summarized in the two presentations and in the specification appeared appropriate, but quite general. In particular, the problems related to each respective site (TMI or Idaho) were emphasized to the exclusion of consideration of the other site or specific details. In contrast, the specification which was based on these considerations attempted to be definitive, but has deficiencies that may adversely affect the canister's ultimate performance.

TAAG has reviewed this information and has the following comments on the proposed specification:

1. The canister design is dependent on the design of the equipment with which it will be used. In particular, the design of the vacuum system and other defueling tools must interface with the canister. These details were not discussed and it is not apparent that these interfaces have received adequate attention.

2. It is not obvious that a single canister design is the optimum solution to the overall fuel canister requirements. It is conceivable that more than one design will be required (different diameters, lengths, complex designs, etc.)

3. The specification does not state that the canister, in conjunction with existing shipping containers, shall constitute a licensable configuration.

4. The possibility of pyrophoric materials in the debris and its impact on canister design is not discussed.

5. The heat transfer considerations of the canister or canister/shipping container configuration are not addressed for long term storage or shipment. It is necessary to show that decay heat is adequately dissipated.

6. The possible consequences of radiological disassociation of the water is limited to the consequences of an explosion or internal pressure buildups. The design requirements for this are not addressed, nor does it address the design implications of disassociation on long term storage.

7. The specification states (Section 3.22) that "the canister shall be vertical for particulate depostion and all subsequent handling". Fuel transfer operations will include a horizontal tranfer of the canister from the canal to the storage racks.

8. The specification (Section 3.16) requires that the canister withstand a vertical longitudinal drop of 60 feet in water with closure caps welded. Deformation is permitted be leakage is not allowed. It is not apparent why this requirement exists. The 60 foot distance is excessive. Considering that the drop occurs within a controlled environment (outside the shipping container) the no leakage limit also appears overly restrictive. A 60 foot drop of a normal irradiated fuel element would most likely not meet these requirements. In addition, it is not obvious that a vertical drop would be most limiting.

9. The specification (Section 3.15) states "there are not specific requirements for an internal pressure design; however . . . ". Pyrophoricity considerations may result in the need for pressurization with an inert gas.

10. The specification states that canisters shall be designed for a minimum 50 year life and the environment shall be a described in Appendix F. Appendix F is not available and can not be evaluated.

11. The specification states that the filter canister top closure shall have an inlet and outlet nozzle of quick disconnect type with check valve. It is not apparent that the benefits of these features will outweigh their "cost" in design completely. Details of available handling concepts are required to evaluate these requirements.

12. The specification in Sections 3.8 and 3.23 specifies the material to be used for the canister and any pipes and valves required for draining the fuel canisters. This detail is not consistent (in excess of) with the degree of guidance provided in the remainder of the specification. It is not apparent why this detail is provided-nor is it obvious that the material selection will be optimum.

13. Section 3.15 indicates that the initial temperature of the canister will be 70°F. Decay heat loads and outside sun/temperature conditions may cause initial canister temperatures in excess of 70°F.

14. A specification should not use terms such as "consider the potential", "the seller shall evaluate the potential", "the seller shall consider the need". Rather the specification should specify what basis should be used to design and construct the canister.

It is suggested that, upon completion of the canister design(s), a prototype(s) canister be purchased. This protoype can be used to confirm the design through a series of test prior to committing to construction the production canisters.

EG&G Idaho, Inc.
PO Box 88, Middletown, PA 17057

February 10, 1983

Distribution

TRANSMITTAL OF FOURTH TAAG REPORT - Hmb-62-83

Dear Sirs:

Enclosed herewith is the fourth report of the Technical Assistance and Advisory Group (TAAG) dated December 1, 1982. This report covers the TAAG activities during the period from September 1 to December 1, 1982.

Very truly yours,

H. M. Burton, Manager
Technical Information & Examination Program

mrr

Enclosure:
As stated

Distribution

TAAG
R. S. Brodsky
D. O. Campbell
N. M. Cole
T. S. Cramer/T. A. Peterson
E. A. Evans
W. H. Hamilton
E. F. Sise, Jr.
E. J. Wagner

U.S. DOE
W. W. Bixby
F. E. Coffman
D. J. McGoff
R. E. Tiller
F. A. Ross

U.S. NRC
L. Barrett
B. Snyder
R. Weller

EG&G Idaho, Inc.
J. A. Dearien, Jr.
T. C. Runion
K. C. Sumpter

GPU Nuclear & Bechtel
R. C. Arnold
J. J. Barton
J. C. DeVine
R. H. Fillnow
R. L. Freemerman
B. K. Kanga
R. L. Rider
H. E. Shaw (6)
J. W. Thiesing

EPRI
J. Taylor
A. Roberts
S. Lefkowitz

Dr. J. C. Fletcher, Univ. of Pittsburgh